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BWRX-300 UK Generic Design Assessment (GDA) Chapter 3 - Safety Objectives and Design Rules for Structures, Systems and Components

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EXECUTIVE SUMMARY

The purpose of this Preliminary Safety Report (PSR) Chapter is to present the general Safety Objectives and Design Rules for Structures, Systems and Components (SSCs) used in the design and assessment of the Boiling Water Reactor (BWR) BWRX-300 reactor design.

The Chapter, along with its Attachment, outlines the general design concepts, requirements, codes and standards applicable for different kinds of SSCs and the approach adopted to meet the safety objectives. The compliance of the actual design with all these elements is demonstrated in further detail in other chapters of the PSR, in particular in those devoted to a description of different SSCs.

The Chapter presents a level of detail commensurate with a 2-step Generic Design Assessment (GDA) and is structured in line with the high-level contents of International Energy Atomic Agency SSG-61.

The safety objectives and design rules presented in the Chapter cover safety functions and functional requirements, radiological acceptance criteria, the Defence-in-Depth (D-in-D) and Defence Lines (DLs) concept and its application, application of general design requirements, and SSCs categorisation and classification. These have been outlined based on international Relevant Good Practice (RGP).

This Chapter and its Attachment, presents relevant information on the design approaches to civil engineering and design of seismic category buildings and structures, mechanical components, instrumentation and control systems, and electrical systems and components.

This Chapter sets out the approach to equipment qualification and an overview of the codes and standards applicable to in-service monitoring, testing, maintenance, and inspections.

Claims and arguments relevant to GDA Step 2 objectives and scope are summarised in Appendix A, along with an As Low As Reasonably Practicable position. Appendix B provides a Forward Action Plan, which includes future work commitments and recommendations for future work where ‘gaps’ to GDA expectations have been identified. UK-specific context information is provided in Appendix C.

ACRONYMS AND ABBREVIATIONS

Acronym	Explanation
AC	Alternating Current
AISC	American Institute of Steel Construction
ALARA	As Low As Reasonably Achievable
ALARP	As Low As Reasonably Practicable
AOO	Anticipated Operational Occurrence
API	American Petroleum Institute
ASCE(/SEI)	American Society of Civil Engineers (/Structural Engineering Institute)
ASME	American Society of Mechanical Engineers
AWWA	American Water Works Association
BDBA	Beyond Design Basis Accident
BPVC	(ASME) Boiler Pressure Vessel Code
BSL	Basic Safety Limit
BSO	Basic Safety Objective
BWR	Boiling Water Reactor
CAE	Claims, Arguments, Evidence
CB	Control Building
CCF	Common Cause Failure
CIV	Containment Isolation Valve
CRD	Control Rod Drive
DBA	Design Basis Accident
DBE	Design Basis Earthquake
DC	Direct Current
DCIS	Distributed Control and Information System
DEC	Design Extension Condition
D-in-D	Defence-in-Depth
DL	Defence Line
DL1	Defence Line 1
DL2	Defence Line 2
DL3	Defence Line 3
DL4	Defence Line 4
DL4a	Defence Line 4a
DL4b	Defence Line 4b
DL5	Defence Line 5
EMC	Electromagnetic Compatibility
EMI/RFI	Electromagnetic/Radio Frequency Interference
EQ	Environmental Qualification

Acronym	Explanation
FMEA	Failure Modes and Effects Analyses
FSF	Fundamental Safety Function
GDA	Generic Design Assessment
GEH	GE Hitachi Nuclear Energy
HVAC	Heating, Ventilation, and Air Conditioning
I&C	Instrumentation and Control
IAEA	International Atomic Energy Agency
IEC	International Electrotechnical Commission
IEEE	Institute of Electrical and Electronics Engineers
INSAG	International Nuclear Safety Advisory Group
ISI	Inservice Inspection
ISRS	In-Structure Response Spectra
IST	In-Service Testing
LfE	Learning from Experience
LV	Low Voltage
LWR	Light Water Reactor
MCR	Main Control Room
MSQA	Management of Safety and Quality Assurance
MV	Medium Voltage
NDE	Non-Destructive Examination
NPP	Nuclear Power Plant
OLC	Operational Limit and Condition
OM	Operations and Maintenance of Nuclear Power Plants
ONR	(UK) Office for Nuclear Regulation
OPEX	Operational Experience
PAM	Post-Accident Monitoring
PIE	Postulated Initiating Event
PSA	Probabilistic Safety Assessment
PSR	Preliminary Safety Report
RB	Reactor Building
RBV	Reactor Building Vibration
RCPB	Reactor Coolant Pressure Boundary
RG	Regulatory Guide
RGP	Relevant Good Practice
RPV	Reactor Pressure Vessel
RRS	Required Response Spectra
SAPs	(ONR) Safety Assessment Principles
SC	Safety Class

Acronym	Explanation
SC1	Safety Class 1
SC2	Safety Class 2
SC3	Safety Class 3
SMR	Small Modular Reactor
SSCs	Structures, Systems and Components
SSE	Safe Shutdown Earthquake
TEMA	Tubular Exchanger Manufacturers Association
UK	United Kingdom
U.S.	United States
USNRC	U.S. Nuclear Regulatory Commission

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REVISION SUMMARY

Revision #	Section Modified	Revision Summary
A	All	Initial Issuance
B	All	Update to reflect changes made across the PSR for end of GDA Step 2 consolidation.
C	All	Update for minor typographical errors

3 SAFETY OBJECTIVES AND DESIGN RULES FOR STRUCTURES, SYSTEMS AND COMPONENTS

Introduction

This chapter presents the safety design basis of the BWRX-300 in a United Kingdom (UK) context. It provides a description of As Low As Reasonably Practicable (ALARP); dose targets and limits; the Defence in Depth (D-in-D) principle and its application; deterministic design principles; equipment qualification and aging. This chapter will also summarise Categorisation and Classification methodology.

This chapter and its Attachment, NEDC-34165P, "BWRX-300 UK GDA Ch. 3: Safety Objectives and Design Rules for SSCs," (Attachment 1) (Reference 3-1), cover general principles and do not include the detailed analyses and substantiation by which each specific area is evaluated.

The chapter presents a level of detail commensurate with a 2-Step Generic Design Assessment (GDA) and is structured in line with the high level contents of International Atomic Energy Agency (IAEA) SSG-61, "IAEA Safety Standards – Format and Content of the Safety Analysis Report for Nuclear Power Plants," (Reference 3-2).

Purpose

This chapter introduces the safety objectives and the safety strategy to meet those objectives for the design and construction of the BWRX-300 in the UK.

Additionally, this chapter describes the methodology for classification of Structures, Systems, and Components (SSCs), the general design aspects, and codes, standards, and RGP applied to the BWRX-300 design to meet the UK regulatory requirements.

Interfaces with Other Chapters

The interfaces between this chapter and other chapters in the Preliminary Safety Report (PSR) are presented as follows:

- NEDC-34166P, "BWRX-300 UK GDA Ch. 4: Reactor (Fuel and Core)," (Reference 3-3), NEDC-34167P, "BWRX-300 UK GDA Ch. 5: Reactor Coolant System and Associated Systems," (Reference 3-4), NEDC-34168P, "BWRX-300 UK GDA Ch. 6: Engineered Safety Systems," (Reference 3-5), NEDC-34169P, "BWRX-300 UK GDA Ch. 7: Instrumentation and Control," (Reference 3-6), NEDC-34170P, "BWRX-300 UK GDA Ch. 8: Electrical Power," (Reference 3-7), and NEDC-34171P, "BWRX-300 UK GDA Ch 9A: Auxiliary Systems," (Reference 3-8) – These chapters present the design of systems and components which are based on the relevant safety and design principles provided in Chapter 3.
- NEDC-34172P, "BWRX-300 UK GDA Ch. 9B: Civil Structures," (Reference 3-9) – presents the design of civil engineering works and structures which is based on the relevant principles presented in Chapter 3.
- NEDC-34173P, "BWRX-300 UK GDA Ch. 10: Steam and Power Conversion Systems," (Reference 3-10) – presents the design of the steam and power conversion systems which is based on the relevant safety and design principles presented in Chapter 3.
- NEDC-34174P, "BWRX-300 UK GDA Ch. 11: Management of Radioactive Waste," (Reference 3-11) – presents the design of systems and components containing radioactive materials based on the safety and design principles provided in Chapter 3.

- NEDC-34175P, "BWRX-300 UK GDA Ch. 12: Radiation Protection," (Reference 3-12) – presents the design of radiation protection based on the safety and design principles presented in Chapter 3.
- NEDC-34176P, "BWRX-300 UK GDA Ch. 13: Conduct of Operations," (Reference 3-13) – uses the relevant engineering substantiation principles presented in Chapter 3 to develop the operational conduct and management for the UK BWRX-300.
- NEDC-34177P, "BWRX-300 UK GDA Ch. 14: Plant Construction and Commissioning," (Reference 3-14) – presents the arrangements and requirements for plant construction and commissioning, considering the relevant principles presented in Chapter 3.
- Chapter 15 – Safety Analysis (References 3-15 through to 3-24) – provides the overarching safety analysis including Probabilistic Safety Assessments (PSAs), Design Basis Analyses (DBAs), and Beyond Design Basis Accidents, including Design Extension Conditions and severe accidents, with the consideration of relevant principles presented in Chapter 3.
- NEDC-34188P, "BWRX-300 UK GDA Ch. 16: Operational Limits and Conditions," (Reference 3-25) – uses the relevant engineering substantiation principles presented in Chapter 3 to develop the operational limits and conditions for the UK BWRX-300.
- NEDC-34189P, "BWRX-300 UK GDA Ch. 17: Management for Safety and Quality Assurance," (Reference 3-26) – presents codes and standards applied to Management of Safety and Quality Assurance (MSQA) which is based on the selection principles of codes and standards in Chapter 3.
- NEDC-34190P, "BWRX-300 UK GDA Ch. 18: Human Factors Engineering," (Reference 3-27) – presents the substantiation of Human Factors principles which are provided in Chapter 3.
- NEDC-34191P, "BWRX-300 UK GDA Ch. 19: Emergency Preparedness and Response," (Reference 3-28) – presents the emergency preparedness and response required by the principles presented in Chapter 3.
- NEDC-34192P, "BWRX-300 UK GDA Ch. 20: Environmental Aspects," (Reference 3-29) – presents the environmental aspects with consideration of the relevant principles presented in Chapter 3.
- NEDC-34193P, "BWRX-300 UK GDA Ch. 21: Decommissioning and End of Life Aspects," (Reference 3-30) – presents codes and guidelines applied in decommissioning and end of life aspects based on the selection principles of codes and standards provided in Chapter 3.
- NEDC-34194P, "BWRX-300 UK GDA Ch. 22: Structural Integrity," (Reference 3-31) – demonstrates the structural integrity by applying design requirements based on the relevant principles presented in Chapter 3.
- NEDC-34195P, "BWRX-300 UK GDA Ch. 23: Reactor Chemistry," (Reference 3-32) – presents codes and guidelines applied in chemistry based on the selection principles of codes and standards provided in Chapter 3.
- NEDC-34196P, "BWRX-300 UK GDA Ch. 24: Conventional Safety and Fire Safety," (Reference 3-33) – presents the applicable codes and standards in conventional safety and fire safety which are compliant with the selection principles of codes and standards provided in Chapter 3.

- NEDC-34197P, "BWRX-300 UK GDA Ch. 25: Security," (Reference 3-34) – describes the general approach to security as well as physical and cybersecurity with consideration of the principles presented in Chapter 3.
- NEDC-34198P, "BWRX-300 UK GDA Ch. 26: Interim Storage of Spent Fuel," (Reference 3-35) – presents applicable codes and standards in interim storage of spent fuel which are based on the selection principles of codes and standards presented in Chapter 3.
- NEDC-34199P, "BWRX-300 UK GDA Ch. 27: ALARP Evaluation," (Reference 3-36) – presents the ALARP evaluation to support and assess the achievement of the nuclear safety objective provided in Chapter 3.
- NEDC-34200P, "BWRX-300 UK GDA Ch. 28: Safeguards," (Reference 3-37) – demonstrates understanding of safeguards requirements at the generic level and how they are accommodated in the standard plant design, with consideration of the principles presented in Chapter 3.

Claims and arguments relevant to the PSR objectives and scope are summarised in Appendix A, along with an ALARP position. Appendix B provides a Forward Action Plan, which includes future work commitments and recommendations for future work where 'gaps' to GDA expectations have been identified. UK-specific context is provided in Appendix C, including UK Context for Numerical Targets which is presented in Section C.1, ALARP context which is presented in Section C.2, and the UK approach to Categorisation and Classification is discussed in Section C.3.

Baseline Design

The BWRX-300 baseline design (Reference 3-38) has been developed and justified in part upon reference to US Nuclear Regulatory Commission (USNRC) guidance and is intended for deployment in the UK. As such, Chapter 3 (and its Attachment) refers throughout to use of USNRC guidance. A Forward Action has been raised to consider alternative codes and standards and justify use as Regulatory Good Practice.

3.1 General Safety Design Basis

The overall safety philosophy for the design of the BWRX-300 is referred to as the Safety Strategy and presented in NEDC-33934P, "BWRX-300 Safety Strategy," (Reference 3-39). 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 based on the principles set forth in the International Atomic Energy Agency document SSR-2/1, "Safety of Nuclear Power Plants: Design," (Reference 3-40).

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

The BWRX-300 design adopts the safety objectives established by the IAEA Safety Standards Series No. SF-1, "Fundamental Safety Principles," (Reference 3-41) and documented in the International Nuclear Safety Advisory Group (INSAG) publication INSAG-12, "Basic Safety Principles for Nuclear Power Plants 75-INSAG-3," (Reference 3-42) 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: To protect individuals, society, and the environment by establishing and maintaining in Nuclear Power Plants (NPPs) an effective defence against radiological hazard

The general nuclear safety objective is supported by the following complementary safety objectives:

- **Radiation Protection Objective**: To ensure in normal operation that radiation exposure within the plant and due to any release of radioactive material from the plant is As Low As Reasonably Achievable (ALARA), economic, and social factors being taken into account, and below prescribed limits, and to ensure mitigation of the extent of radiation exposure due to accidents.
- **Technical Safety Objective**: To prevent with high confidence accidents in nuclear plants; to ensure that, for all accidents taken into account in the design of the plant, even those of very low probability, radiological consequences, if any, would be minor; and to ensure that the likelihood of severe accidents with serious radiological consequences is extremely small.

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

As Low As Reasonably Practicable

It is necessary to show that the risks to the workers and the public are ALARP. This requires that all reasonable measures are taken in the design, construction, and operation of the plant to minimize the radiation dose received by workers and public, unless such measures are grossly disproportionate to the risk avoided.

The ALARP methodology and evaluation are provided in NEDC-34199P (Reference 3-36).

Appendix C presents further discussion on the legal basis for ALARP in the UK and the approach to ALARP that will be presented in this PSR.

3.1.2 Fundamental Safety Functions

NEDC-33934P (Reference 3-39) defines and maintains the Fundamental Safety Functions (FSFs) to ensure protection of the physical barriers. For a given event sequence, if the functional DLs required to fulfill the FSFs are performed successfully, then the corresponding barriers remain effective.

The design of the BWRX-300 fulfills FSFs at all plant states (defined in Subsection 3.1.5) which ensures the design meets its safety objectives. 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 FSFs are defined in IAEA SSR-2/1 (Reference 3-40).

A systematic approach is taken to identify the FSFs and those SSCs necessary to fulfill the FSFs following a Postulated Initiating Event (PIE). The results of applying the systematic approach are gathered in the fault list, which provides traceability between DL functions with a direct role in fulfilling an FSF and the plant states and event sequences in which each of those functions performs that role.

Fulfillment of the FSFs 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 fulfillment of the FSFs. Other considerations for the monitoring function are as follows:

- 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.
- 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.
- 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.
- A minimum set of monitoring functions and display of parameters that do not support the operator actions are provided to support accident assessment.

Fulfillment of the FSFs is intrinsic to BWRX-300 Safety Strategy. A systematic approach is taken to identify the FSFs and those SSCs necessary to fulfill the FSFs following a PIE. This systematic approach is detailed in the D-in-D discussion in Subsection 3.1.7.

3.1.3 Radiation Protection and Radiological Acceptance Criteria

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.

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 NEDC-34190P (Reference 3-27) 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 NEDC-34175P (Reference 3-12).

PSR Subchapter 15: Safety Analysis (Including Fault Studies, PSA, and Hazard Assessment," (Reference 3-15) describes the dose calculation methodology used in the deterministic safety analysis. Results of the analyses are summarized in PSR Subchapter 15.9: Safety Analysis: Summary of the Results of the Safety Analyses (Including Fault Schedule)," (Reference 3-24) demonstrating that the radiological consequences of the analysed events do not exceed the acceptance criteria for Anticipated Operational Occurrences (AOOs) and for DBAs.

3.1.4 Safety Goals

In addition to the dose acceptance criteria, 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.

For the BWRX-300 standard plant design these plant safety goals are presented in NEDC-33934P (Reference 3-39) and reproduced below:

- **Core damage frequency** - The sum of frequencies of all fault sequences that can lead to significant core degradation shall be less than 1E-6 per reactor-year
- **Large release frequency** - The sum of frequencies of all fault sequences that can lead to a release to the environment that requires long-term relocation of the local population shall be less than 1E-7 per reactor-year

The PSA is described in detail in PSR Subchapter 15.6: Safety Analysis: "Probabilistic Safety Assessment", (Reference 3-21).

Appendix C presents further discussion of these safety goals in the context of UK numerical targets for normal operational, design basis fault and radiological accident risks to people on and off the site.

3.1.5 Plant States Considered in the Design Basis

The range of conditions and events considered are categorised 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:

- Normal Operation includes the operational states that are expected to occur frequently or regularly during plant operation, including the following Normal Plant Operational Modes: Power Operation, Startup, Hot Shutdown, Stable Shutdown, Cold Shutdown, and Refuelling, maintenance, or manoeuvring of the plant (the normal plant operating modes are described in NEDC-34188P (Reference 3-25)).
- AOOs 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 classified 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 (BDBAs), 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 representative 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 3.1.9.

Events are assigned to a plant state based on the expected frequency of the fault sequence, which includes a PIE and, in some cases, additional failures of mitigating functions. 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 SSCs 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 NEDC-34178P (Reference 3-15).

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 NEDC-34176P Reference 3-13 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 Operating Limits and Conditions (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 NEDC-34188P (Reference 3-25).

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 PSR Subchapter 15.3: Safety Analysis: "Safety Objective and Acceptance Criteria", (Reference 3-18).

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 3.1.4) (PSA is described in detail in NEDC-34178P, (Reference 3-15)). 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 NEDC-34176P (Reference 3-13) and through complementary design features described in NEDC-34178P (Reference 3-15).

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 categorising mitigating functions based on their ability to prevent and mitigate the progression of events ensuring that the safety objectives are met.

3.1.6 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 D-in-D. The application of D-in-D for the BWRX-300 design is described below.

3.1.7 Defence-in-Depth

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.

3.1.7.1 BWRX-300 Defence-in-Depth Concept

The concept of D-in-D involves the provision of multiple layers of defence against some undesirable outcome rather than a single, strong defensive layer. In the case of a NPP, 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 “DLs”. 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 DLs without explicit acknowledgment of the physical barriers. The fundamental purpose of the DLs 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 DLs and the physical barriers. In a given plant scenario, if the FSFs are performed successfully, then the corresponding physical barriers remain effective.

3.1.7.2 Defence Lines

Five DLs (or levels), DL1 through Defence Line 5 (DL5), are adopted consistent with IAEA SSR-2/1 (Reference 3-40). Figure 3-1 illustrates the DLs as they correspond to the plant states.

The first 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 DLs by assuring high quality and conservatism in design, construction, and operation. The second, third, and fourth DLs (DL2, DL3, and DL4) comprise plant functions that act to prevent PIEs from leading to significant radioactive releases. The fifth DL5 involves off-site emergency preparedness to protect the public in case a substantial radioactive release occurs.

The DLs 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 DLs 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 provides a high-level description of the objective, and the design means and operational means for supporting the DLs. The following is a brief description of each of the DLs.

3.1.7.2.1 Defence Line 1

The purpose of the first level of defence is to prevent deviations from normal operation and the failure of important SSCs. 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 NEDC-34183P, "BWRX-300 UK GDA Ch. 15.5: Safety Analysis: Deterministic Safety Analyses," (Reference 3-20).

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 SSCs
- 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
- Categorisation and qualification of SSCs according to their safety significance
- Operational Limits and conditions
- Application of lessons learned through operating experience

3.1.7.2.2 Defence Line 2

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 Defence Line 2 (DL2).

Examples of DL2 measures include:

- Anticipatory plant trips
- Maintain target power
- Maintain target level
- Maintain target pressure
- Control Rod Block

3.1.7.2.3 Defence Line 3

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. Defence Line 3 (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 NEDC-34183P (Reference 3-20).

Examples of DL3 measures include:

- Reactor Scram
- Isolation Condenser Initiation
- Main Steam Isolation
- Containment Isolation
- Reactor Pressure Vessel (RPV) Isolation

3.1.7.2.4 Defence Line 4

The purpose of the fourth level of defence is to mitigate DECs.

For the BWRX-300, Defence Line 4 (DL4) is comprised of two subsets of functions that are designated as Defence Line 4a (DL4a) and Defence Line 4b (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 SSCs from performing their intended function (i.e., Common Cause Failure (CCF) which is a failure of two or more SSCs 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 SSCs 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 NEDC-34178P (Reference 3-15).

3.1.7.2.5 Defence Line 5

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.

Defence Line 5 (DL5) includes emergency preparedness measures to cope with potential unacceptable releases in case the first four DLs are not effective. These are largely off-site measures taken to protect the public in a scenario involving substantial release of radiation.

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). NEDC-34191P (Reference 3-28) discusses emergency response arrangements such as procedures and facilities. Communication systems are discussed in NEDC-34171P (Reference 3-8) (note that these measures may be initiated earlier in an event prior to progression to a severe accident).

3.1.7.3 Defence Line Independence

The BWRX-300 design incorporates independence in the application of D-in-D. DLs 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:

- Among DL2, DL3 and DL4a, at least one DL can mitigate a PIE caused by or concurrent with a CCF in another DL, with the mitigation means being independent from the effects of the initiating CCF.
- All PIEs with a frequency greater than 1E-05 per reactor year 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.
- The DL4b functions intended for mitigating DECs are functionally and physically separated from the systems intended for other DL functions.
- 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.
- Exceptions to rules of independence are described, assessed, and justified. If equipment supports functions in more than one DL, there is an increased focus on their

reliability in the application of DL1 compared to a design feature credited in only one DL.

3.1.7.4 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 3-2. These include:

- Hazard Evaluations
- Fault Evaluation
- Deterministic Safety Analyses
- PSA

The elements of Figure 3-2 are briefly described below.

3.1.7.4.1 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 PSR Subchapter 15.1: Safety Analysis: “General Considerations”, (Reference 3-16):

- Functional Failure Hazard Evaluation – assessment of failures of SSCs
- 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.

3.1.7.4.2 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).

The Fault Evaluation establishes traceability between the plant design and the safety analysis bases. The Fault Evaluation process including the selection and categorisation of PIEs and fault sequences for deterministic safety analysis is described in PSR Subchapter 15.2: Safety Analysis: “ID, Categorisation and Grouping of PIEs and Accident Scenarios”, (Reference 3-17).

3.1.7.4.3 Deterministic Safety Analyses

The objective of deterministic safety analysis for NPPs is to confirm that:

- FSFs can be performed
- SSCs 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 analysed in deterministic safety analysis. NEDC-34183P (Reference 3-20) provides more detail on the deterministic safety analysis process.

3.1.7.4.4 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 Subsection 3.1.4 – Safety Goals) 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 modelled 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 NEDC-34184P (Reference 3-21).

3.1.8 Application of General Design Requirements and Technical Acceptance Criteria

3.1.8.1 Deterministic Design Principles in Codes and 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 SSCs 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.

3.1.8.2 Minimise Probability of Structures, Systems and Components Failure

The probability of failure of systems and equipment is minimised through a design which provides predictable and repeatable performance of the FSFs. This is achieved by deploying highly reliable and dependable SSCs.

DL3 systems and equipment are designed to fail to a safe state or to a known, defined state to ensure safety is not jeopardised. Thus, scram 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.

Integration of software into the overall system development process is a fundamental aspect of minimising failure probability. The Instrumentation and Control (I&C) System Life Cycle is applied to each I&C system which follows the overall lifecycle presented in International Electrotechnical Commission (IEC) 61513, “Nuclear Power Plants – Instrumentation and Control Important to Safety – General Requirements for Systems,” (Reference 3-43). Further details on this process are provided in NEDC-34169P (Reference 3-6), 006N2631, “I&C Plant Level Design Assurance Plan,” (Reference 3-44), and 006N9508, “BWRX-300 Program Configuration Management Implementation Plan,” (Reference 3-45).

NEDC-34176P (Reference 3-13) describes how fitness for service is addressed in established programs that include: Reliability, Maintenance, Aging Management, Chemistry Control, Periodic and Inservice Inspections (ISIs). Programmatic requirements addressing fitness for service span the full life cycle of the facility beginning with inclusion in facility design decision making.

3.1.8.3 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 SSCs 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 Subsection 3.1.7 (Safety Strategy Process for Implementing D-in-D) and described in more detail in NEDC-34180P (Reference 3-17) and confirmed via the PSA in NEDC-34184P (Reference 3-21).

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 3.1.8 (Single Failure Criterion).

3.1.8.4 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 DLs.

3.1.8.5 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 DL function independence discussed in Subsection 3.1.7 (DL Independence). 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 SSCs that perform DL functions are co-located with other plant equipment that do not.

3.1.8.6 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.

3.1.8.7 Single Failure Criterion

In the BWRX-300 design, reliability targets are established for the functions in each defense line. The systems implementing these functions incorporate redundancy to the extent required to satisfy these reliability targets. Additionally, the single failure criterion, which promotes reliability through deterministic application of redundancy, is adopted for DL3 functions. Each

DL3 function is successfully performed in the presence of any single component failure in any single system that supports performance of the function (excluding components whose failure is ‘practically eliminated’). Compliance with this criterion is a precondition to achieving the DL3 function reliability target.

The Single Failure Criterion is applied in two distinct ways:

- Specific analyses are conducted, which perform systematic searches for single failure vulnerabilities in systems that perform DL3 functions (e.g., Failure Mode and Effects Analyses). This approach demonstrates that the architecture of those systems supports the single failure claim.
- In the Conservative Deterministic Safety Analysis, the most challenging single failure which could occur in conjunction with a given PIE is assumed to occur in the safety analysis performance model. This approach confirms that performance objectives are met in a manner that supports the single failure claim.

In applying the criterion, single failures of both active components (e.g., a valve fails to change state on demand) and of passive components (e.g., a break in a pipe) are considered for their potential to challenge performance of Fundamental Safety Functions (FSF) through the Fault Evaluation.

Design Rules that reflect the single failure criterion are applied to systems and equipment which implement the DL3 functions. Although the criterion is not formally applied to equipment performing DL2, DL4a, or DL4b functions, redundancy is applied to the design of those systems as needed to achieve the associated reliability targets.

3.1.8.8 Common Cause Failures

3.1.8.8.1 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 (SC) 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 minimise 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 D-in-D, and it increases the chances that a DL function will be available when needed. Different DLs 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 SSCs. Quality assurance and quality control measures applied to SSCs commensurate with their importance help reduce preclude potential CCFs.

3.1.8.8.2 Common Cause Failures of Digital Instrumentation and Control Software

The BWRX-300 approach to assessment of CCF of Digital 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 modelling 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. Analysing 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.

3.1.8.8.3 Defence Line Approach to Common Cause Failure

A multi-pronged approach and the systematic integration of CCFs in DL 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:

- 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.
- For internal and external events resulting in DECs, the design includes independent and diverse system functions to cope with the effects of CCF (e.g., DL4a).
- Diverse accident monitoring instrumentation for severe accident management (e.g., DL4b) is provided.

The D-in-D 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.

3.1.8.9 Other Approaches for Ensuring Safety

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

3.1.8.9.1 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 SSCs, simplifies operations and maintenance.

3.1.8.9.2 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:

- Safety Class 1 (SC1) batteries are capable of powering loads for a minimum of 72 hours. The design ensures that plant safety is maintained even after battery depletion.
- The BWRX-300 design utilises passive natural circulation for fuel cooling and 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 utilisation 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.

Radiological Protection Principles

Administrative programs and procedures, in conjunction with facility design, ensure that occupational radiation exposure to personnel is kept ALARP. The systematic application of the ALARP principle during the design phase of the BWRX-300 establishes the basic design criteria observed to reducing occupational exposure during plant operation and maintenance, decommissioning and post-accident ALARP.

ALARP design requirements are established to improve the layout of enclosures, accesses, and exits from controlled areas of the plant structures that confine radioactive material. The design of plant SSCs minimizes personnel exposure to radiation during operation, inspection, maintenance, or plant design modifications.

The ALARP design requirements keep radiation exposures ALARP during normal operation or AOOs and planned radioactive material releases below regulatory limits. The ALARP design criteria includes provisions for mitigating the radiological consequences of design basis accidents.

The BWRX-300 plant design:

- Precludes the release of radioactive material to the public and the environment that exceeds the limits of applicable regulations for normal operations, transients, and accidents
- Minimises personnel exposure
- Minimises the generation of radioactive contamination and waste

The following BWRX-300 design features minimise radioactive contamination:

- Containment in areas where leaks and spills are likely to occur
- Leak detection capability to provide prompt SSCs leakage
- Usage of leak detection methods (e.g., instrumentation, automated samplers) capable of early leak detection in areas difficult (inaccessible) to conduct regular inspections (such as the fuel pool), and buried, embedded or subterranean piping) to avoid release of contamination. All BWRX-300 tanks containing radioactive fluids are within the Radwaste Building that have cubicles and drain back into the radioactive liquid waste for processing.
- Minimizing embedded piping, sumps, or buried equipment to facilitate decommissioning
- Removal or replacement of equipment or components during facility operation or decommissioning
- Minimizes the generation of radioactive contamination and waste during operation decommissioning by reducing the volume of components and structures that become contaminated during plant operation

Design for Decommissioning Principles

Operational Experience (OPEX) demonstrates that decommissioning of reactor facilities is best facilitated if considered during the design phase. Assessment of future facility decommissioning and dismantling activities at the design phase include consideration of OPEX gained from the decommissioning of existing facilities, as well as those facilities that are in long-term safe storage. The consideration of decommissioning at the design phase is expected to result in lower worker doses, reduced environmental impacts, and improved life cycle management of the facility.

BWRX-300 design features to facilitate decommissioning include:

- Optimised for constructability, which may be beneficial for dismantling the facility during decommissioning
- Modularisation which will provide guidance in selection of disassembly methods employed during decommissioning
- Maintaining low occupational exposures

- Provisions for draining, flushing, and decontaminating equipment and piping
- Design of equipment to minimise the buildup of radioactive material and to facilitate flushing of piping systems
- Separation of more highly radioactive equipment from less radioactive equipment

NEDC-34193P (Reference 3-30) provides further details on design for decommissioning.

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 NEDC-34181P (Reference 3-18). Details of the deterministic safety analysis are presented in NEDC-34178P (Reference 3-15).

3.1.9 Practical Elimination

Consistent with IAEA SSR-2/1 (Reference 3-40), 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 3-40) 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 representative cases or type of accident conditions.

Severe accident phenomena based on operating experience with predecessor advanced Light Water Reactors (LWRs) serve as a starting point for consideration for practical elimination.

Analyses demonstrating practical elimination are described in NEDC-34178P (Reference 3-15).

3.1.10 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.

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 NEDC-34183P (Reference 3-20). As part of the PSA, sensitivity and uncertainty analysis is conducted to demonstrate consideration of potential cliff-edge effects (See PSR Subchapter 15.6, Reference 3-21).

3.1.11 Design Approaches for the Reactor Core and for Fuel Storage

3.1.11.1 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.

3.1.11.2 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 SSCs that are necessary to fulfil 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
- 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
- To initiate appropriate safety actions

Refer to NEDC-34171P (Reference 3-8) for a detailed description of the Fuel Handling and Storage Systems.

3.1.12 Consideration of Interactions Between Multiple Units

The scope of UK GDA is for a single unit. However, interactions between multiple units has been considered as detailed below.

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 SSCs, 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.

Each BWRX-300 unit would have its own SC 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.

3.1.13 Design Considerations for Aging Management

Aging of SSCs 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 Subsection 3.9.

In designing components, system designers consider aging mechanisms and their effects on the safety, reliability, and performance of SSCs 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 license renewal for operating plants.

Known aging phenomena are quantified and considered in the design of SSCs. 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. Analysed 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:

- A recommended strategy for aging management and prerequisites for its implementation
- Identification of SC SSCs in the plant that could be affected by aging.
- Proposals for appropriate materials monitoring and sampling programs, where aging may affect the capability of critical SSCs to perform their functions throughout the lifetime of the plant
- Appropriate consideration of operating experience with respect to aging
- Recommendations for aging management for SC SSCs (RB Diaphragm Plate Steel-Plate Composite (DP-SC) structures, mechanical components, electrical and instrumentation and control components, cables, etc.) and measures to monitor and mitigate their degradation
- Equipment qualification requirements of SC SSCs

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

SSCs, radiation shielding, damping of vibrations, submerged conditions and water chemistry, selection of cable routes, etc.).

3.2 Categorisation of Functions and Classification of Structures, Systems and Components

The BWRX-300 approach to categorisation of functions and classifying SSCs is consistent with IAEA SSR-2/1 (Reference 3-40) and IAEA SSG-30, "Safety Classification of Structures, Systems and Components in Nuclear Power Plants," (Reference 3-46). Classification of SSCs is conducted to identify the importance of the SCC with respect to safety.

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

- SC
- Seismic Category
- Quality Group

Classification of SSCs 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 SSCs. Subsections 3.6 through 3.9 in Attachment 1 of NEDC-34165P (Reference 3-1) describe the codes and standards applicable to civil, mechanical, I&C, and electrical SSCs based on classification.

The classification of SSCs also determines the degree of redundancy, diversity, separation, and reliability/availability required as described in Subsection 3.1.8. The requirement for Environmental Qualification (EQ) is based on the classification of SSCs as described in Subsection 3.9.3. In addition, SSCs classification informs procurement and quality assurance requirements as discussed in NEDC-34189P (Reference 3-26).

Appendix C provides further discussion on the BWRX-300 approach to categorisation of functions and classifying SSCs in the context of UK expectations.

3.2.1 Safety Classification Background

The BWRX-300 approach to classifying SSCs is based primarily on deterministic methods and is directly traceable to the safety functions performed by the SSCs. This approach reflects:

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

A fundamental element of the BWRX-300 SSCs classification approach is the direct correlation between the DL in which an SSCs performs a function, and the relative safety importance of that function.

3.2.1.1 Primary Function Categorisation

Section 4.2 of NEDC-33934P (Reference 3-39) provides a description of the process for assigning Safety Categories to Functions and 005N9461 (Reference 3-100) discusses the reliabilities of the defence line functions. The process for assigning safety categories to functions is illustrated in NEDC-33934P, Table 4-1: Functional Safety Category Assignment NEDC-33934P (Reference 3-39), and identifies the SSCs functions that apply to each Safety Category for the following groups of functions:

- DL2 Functions (10^{-2} failures per demand) – Actively control key plant parameters associated with FSFs and detect and mitigate AOO PIEs
- DL3 Functions (10^{-4} failures per demand) – Detect and mitigate DBA PIEs and event sequences comprising AOO PIEs and failure of DL2 functions

- DL4a Functions – Detect and mitigate DECs, including event sequences associated with some DBA PIEs and failure of DL3 functions
- DL4b Functions – Detect and mitigate DECs to prevent core damage or mitigate the consequences of core damage events (severe accidents)
- Normal Functions – Functions typically operating during normal plant operation
- PAM Functions – Support monitoring and display of PAMs variables

Primary functions are those that directly perform the FSFs in support of DL2, DL3, DL4a or DL4b. Functions are categorised into three safety categories: Safety Category 1, Safety Category 2, and Safety Category 3, with Safety Category 1 being the most important. Safety Categories are applied to the primary functions as follows:

- 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.
- 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.
- Safety Category 3 is assigned to DL2 and DL4b primary functions as they are the least important to safety. 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 the least important to safety DL functions, despite the high consequence of failure.
- Non-Safety Category is assigned to all other functions.

In addition to categorising primary functions by the DL 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 summarised in Table 3-2.

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., a Heating, Ventilation, and Air Conditioning (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 DL function (and therefore subject to DL 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 DL function (and therefore not subject to DL function "rules," such as independence and diversity). The primary function would generally 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, it can be classified as Safety Category 2 (instead of Safety Category 1), and if the SSCs are 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 DL function "rules," such as independence and diversity.

3.2.1.5 Normal Functions

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

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, 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.

Some component classifications are made for components that perform FSFs but may not be explicitly defined as part of a DL 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 whose structural failure could damage the fission product barriers are assigned to SC1.
- Components that are part of the RCPB are assigned to SC1.
- Structures (excluding fuel handling equipment) are assigned a safety classification based on the highest safety classification of the components they house or support, excluding components whose failure, due to failure of the structure, results in fail-safe performance of the component’s safety category functions.

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 SC as the system itself. For example, a process system may be classified as SC1 because one or more of its components support a DL3 function; however, the system may also contain components that support functions associated with other DLs or components that support no DL functions. These components are classified in accordance with the DL functions they support.

Structures are assigned a safety classification based on the highest safety classification of the components they house or support. 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) need not be considered in the classification of the structure. The seismic categorisation of a building drives the design rules and performance requirements associated with preventing and mitigating the effects of external and internal hazards. Seismic categorisation methodology is described in Subsection 3.2.3.

3.2.2 Safety Classification Process

In alignment with IAEA guidance, this method of classifying the safety significance of SSCs 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 systems and components based on their safety classification and the DL functions they support. Design bases for structures are derived from their seismic category. 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 Subsection 3.1.7 – Hazard Evaluations) to identify hazards with potential to challenge an FSF. The output of these hazard evaluations are potential PIEs.

Grouping and Identification of Representative PIEs – Potential PIEs are grouped by plant effect and occurrence frequency. Representative PIEs and fault sequences are selected for deterministic safety analyses as described in NEDC-34180P (Reference 3-17).

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 DL is carried out in the Fault Evaluation described in NEDC-34180P (Reference 3-17) with traceability of each function to each PIE and PIE sequence in which it is credited.

Safety Categorisation of the Safety Functions – Assignment of a function designed to mitigate one or more PIEs to a DL reflects its relative importance to safety and its role in maintaining the FSFs under off-normal conditions. As such, each function receives a safety categorisation directly based on its assignment to a DL (as described in Subsection 3.2.1 above). 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

Identification of SSCs 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 modelled 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 SSCs to a Safety Class Corresponding to the Safety Category – SC is assigned to SSCs based on the SSCs' safety category.

Verification of SSCs Classification – The deterministic safety analyses are maintained and updated as the plant design matures. Confirmation of SSCs 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 SSCs – Engineering design rules are applied to SSCs 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 SSCs, as well as requirements for reliability, diversity, redundancy, and independence applicable to an SSCs. These design rules are discussed in Subsection 3.1.8.

3.2.3 Seismic Categories

Seismic Category reflects SSCs requirements during and after a seismic event and governs how the SSCs is seismically designed and qualified. BWRX-300 Seismic Category is assigned as follows:

- **Seismic Category 1A or 1B** – SSCs that are required to remain functional during and after a seismic event are considered Category 1A or 1B:
    - Seismic Category 1A for passive structures and components that are required to remain structurally intact
    - Seismic Category 1B for active components that are required to remain structurally intact and functional
- **Seismic Category 2** – SSCs that are not required to remain functional during or after a seismic event, but whose failure during a seismic event could adversely affect the ability of any Seismic Category 1A or 1B SSCs to accomplish its Safety Category 1 function.

- **Seismic Category RW -** SSCs for management and storage of radiological material that meet the criterial for RW-IIa (High Hazard) in U.S. Nuclear Regulatory Commission (USNRC) Regulatory Guide (RG) 1.143 “Design Guidance for Radioactive Waste Management Systems, Structures and Components Installed in Light-Water-Cooled Nuclear Power Plants”, (Reference 3-47) are classified as Seismic Category RW. RG 1.143 permits the use of the ASCE/SEI 43, “Seismic Design Criteria for Structures, Systems and Components in Nuclear Facilities,” (Reference 3-48) graded approach for the seismic classification of SSCs with justification. These SSCs are therefore designed to remain essentially elastic without any significant permanent deformation up to half of the Design Basis Earthquake (DBE). The use of the ½ DBE is justified as it bounds the ground motion spectra for seismic categories identified in ASCE/SEI 43 (Reference 3-48) for SSCs used for handling and storage of highly radioactive materials.
- **Seismic Category NS** - All other SSC are categorised as Non-Seismic (NS) and are designed based on applicable non-nuclear requirements.

See PSR Subchapter 15.8: Safety Analysis: “External Hazards”, (Reference 3-23) for further discussion on the application of the one-half site-specific DBE approach for BWRX-300 Seismic Category RW SSCs in the UK.

The BWRX-300 Containment and the Reactor Building (RB) are the only structures that house, support, or protect BWRX-300 SC1 SSCs. These two structures are therefore categorised as BWRX-300 Seismic Category 1A structures in the BWRX-300 design.

3.2.3.1 Seismic Interaction

SSCs that are not BWRX-300 Seismic Category 1A or 1B, but whose failure during a seismic event could adversely affect the ability of any Seismic Category 1A or 1B SSCs to accomplish its safety function, are evaluated for seismic interaction to demonstrate that:

- These SSCs will not collapse or collide with the BWRX-300 Seismic Category 1A or 1B SSCs and will maintain their stability during a DBE or other relevant extreme external hazard event
- Impact loads that result from collapse or collision on the BWRX-300 Seismic Category 1A or 1B SSCs are either negligible or smaller than those considered in the design

Interaction evaluations are performed of the Power Block structures and foundations adjacent to the RB to ensure:

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

3.2.4 Quality Group

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 SC and is adequate to provide confidence that plant failures are minimized.

BWRX-300 design utilises a Quality Group designation per the guidance in USNRC RG-1.26, “Quality Group Classifications and Standards for Water-, Steam-, and Radioactive-Waste-Containing Components of Nuclear Power Plants,” (Reference 3-49) 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 The guidance and classification method are used with some clarification based on the unique design of the BWRX-300.

One exception is taken to the guidance in RG 1.26 with respect to Reactor Isolation Valves, as initially discussed in NEDC-33911P-A, "BWRX-300 Containment Performance". Reactor Isolation Valves that function as the inboard Containment Isolation Valves (CIVs) are designed in accordance with the rules and requirements of American Society of Mechanical Engineers (ASME) Boiler and Pressure Vessel Code (BPVC), Section III, Division 1, Subsection NB, Class 1 Components.

Table 3-3 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 3-49), appropriate industrial codes and standards are applied.

3.3 Protection Against External Hazards

The BWRX-300 design considers natural and human-induced external hazards that may be linked with significant radiological risk. This section discusses external hazards and the BWRX-300 approach to prevent and mitigate their effects on SC1 SSCs. SC2/SC3 SSCs that are credited in the fault evaluation with mitigating fault sequences initiated by external hazards, and SSCs whose failure can affect the structural integrity or SC functions of adjacent SC1 SSCs 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 NEDC-34196P, "BWRX-300 UK GDA Ch. 2: Site Characteristics," (Reference 3-56). 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 DBA or a 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 SSCs do not fail when subject to these hazards and generated loadings. Demonstration of the adequacy of protection measures is provided in the applicable PSR chapters covering the design of SSCs.

Malevolent acts considered in the robustness design are discussed in Subsection 3.3.3 of Attachment 1 in NEDC-34165P (Reference 3-1), (Other External Hazards – Robustness Against Malevolent Acts).

Protection and mitigation methods considered in the design are in line with the design safety objectives and D-in-D concept discussed in Subsections 2.1 and 2.6, 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. 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 PIEs with a potential to challenge a fundamental safety function, and to organize them into the fault list developed as per NEDC-34178P (Reference 3-15). Combinations of randomly occurring individual events are considered in these evaluations. Deterministic and probabilistic safety analyses are then performed as discussed in NEDC-34183P (Reference 3-20) and PSR Subchapter 15.6 (Reference 3-21), to confirm the design adequacy and its resilience to these hazards.

See Section 3A.3 of NEDC-34165P Attachment 1 (Reference 3-1) for further detail on protection against external hazards.

3.4 Protection Against Internal Hazards

This section discusses design basis internal hazards that could compromise the safety functions of SC1 SSCs and preventive, and mitigation measures implemented in the design to eliminate their adverse effects. SC2/SC3 SSCs 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 NEDC-34178P (Reference 3-15).

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. Screening methodology (see Reference 3-22) 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 Attachment 1, Section 3A.3 of NEDC-34165P (Reference 3-1).

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 3.1.7, respectively. They include the use of separation, barriers/shielding and monitoring programs as described in Subsection 3.1.2 to preclude unacceptable radiation releases following accidents due to internal hazards.

Combination of loads from randomly occurring individual internal hazards is also considered in the design to ensure structure are adequately protected against internal hazards.

See Section 3A.4 of NEDC-34165P, Attachment 1 (Reference 3-1), for further detail on protection against internal hazards.

3.5 Design of Civil Structures

Section 3A.5 of NEDC-34165P, Attachment 1 (Reference 3-1) presents the general design principles, general design basis requirements and general criteria used in the design of the BWRX-300 civil structures, including their foundations.

3.6 Mechanical Systems and Components

Subsection 3A.6 of NEDC-34165P Attachment 1 (Reference 3-1) provides the general design aspects used for SC and Non-Safety Class (SCN) mechanical systems and components. It includes special considerations for mechanical components, dynamic testing, and analysis of SSCs, required codes for ASME BPVC Section III, Division 1, Class 1, 2, and 3 components, and Subsection NF for component supports, and Subsection NG for core support structures. In addition, general design aspects for Control Rod Drive (CRD) system, and reactor vessel internals are presented. Further, this section discusses the functional design, qualification, and In-Service Testing (IST) program requirements for pumps, valves, and dynamic restraints.

The general design principles, criteria, and classification used for design of mechanical systems and components have been discussed earlier in PSR Chapter 3. 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 that is commensurate with the safety classification and seismic category. The design and qualification of mechanical components is performed using a graded approach with the highest level of rigor applied to SC1 components.

Subsection 3A.6 in NEDC-34165P Attachment 1, (Reference 3-1) also develops the seismic input criteria and building spectra used as input for seismic qualification of Seismic Category I active mechanical components and system functionality. Additionally, Seismic Category I passive mechanical component supports, and equipment supports use the seismic spectra for qualification.

Equipment qualification requirements are provided in Subsection 3.9 of this Chapter for seismic and dynamic qualification of mechanical and electrical equipment, and provides the equipment qualification requirements including environmental, functional qualification, and Electromagnetic Compatibility (EMC), which are used as input to safety classified mechanical systems and components.

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 DLs. The DCIS provides control, monitoring, alarming and recording functions. Although normally integrated, the various components of the DCIS are designed to operate independently.

See Subsection 3A.7 in Attachment 1 of NEDC-34165P (Reference 3-1) for further discussion on the general design aspects, and NEDC-34169P (Reference 3-6) for further detail on the I&C systems and components.

3.8 General Design Aspects for Electrical Systems and Components

The electrical power system design is a 50 Hz Alternating Current (AC) power system, with 6.9 kV for the Medium Voltage (MV) level and 690 VAC (Volts Alternating Current), and 400/230 VAC for the Low Voltage (LV) level.

The BWRX-300 design minimises the reliance on electrical power to support safety category functions. The passive design of the plant is not dependent upon AC power sources including diesel generators, to mitigate a DBA. SC1 power is supplied from battery-backed Direct Current (DC) power, which has a coping period of 72 hours for all DBAs.

See Subsection 3A.8 in Attachment 1 of NEDC-34165P (Reference 3-1), for further discussion on the general design aspects, and NEDC-34170P (Reference 3-7) for further detail on the electrical systems and components.

3.9 Equipment Qualification

3.9.1 Introduction

3.9.1.1 Purpose

Equipment qualification is the process carried out (including the generation and maintenance of evidence) to ensure SSCs 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.

3.9.1.2 Scope

Equipment qualification requirements are applied to BWRX-300 equipment based on the assigned safety classification and seismic categorisation of SSCs (as described in Subsection 3.2.3), and to certain post-accident monitoring equipment.

Equipment qualification considers all normal operating conditions in which the SSCs are expected to operate including conditions arising from maintenance and testing, and also, the conditions arising from AOOs, DBAs, and internal and external hazards.

DEC survivability assessments are outside the scope of a qualification program. However, IEC/IEEE 60780-323, "Nuclear facilities – Equipment important to safety – Qualification," (Reference 3-57) considers qualifying equipment for DEC and the guidance IEC/IEEE 60980-344, "Nuclear facilities – Equipment important to safety – Seismic qualification," (Reference 3-58) can be used to demonstrate with reasonable confidence, that SSCs will survive and perform their intended fundamental safety function(s) under the expected conditions for the timespan required.

3.9.1.3 Aging Considerations

Significant aging mechanisms are considered in establishing EQ 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 SC 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. Age conditioning considers sequential, simultaneous, and synergistic effects to achieve the worst state of degradation.

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.

3.9.2 Seismic and Dynamic Qualification of Mechanical and Electrical Equipment

The BWRX-300 Seismic Category 1A or 1B (hereafter referred as Seismic Category I) mechanical and electrical equipment (including I&C components) are designed to withstand the effects of earthquakes (i.e., Seismic Category I requirements), and other accident-related dynamic loadings.

Mechanical equipment consists of items of a facility including pumps, valves, valve operators, vessels, and piping whose function is required to ensure safe operation or safe shutdown. Electrical equipment consists of all electrical power and 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. Examples of electrical equipment are battery and battery racks, instrument and instrument racks, control consoles, electrical cabinets, electrical panels, valve operator motors, solenoid valves, pressure switches, relays, level transmitters, electrical penetrations, and pump and fan motors.

SSCs that are credited to remain functional during or after a seismic event are Seismic Category I. SSCs whose failure during a seismic event could adversely affect the ability of Seismic Category I SSCs to accomplish their fundamental safety functions are considered for the qualification.

Subsection 3.9.2 addresses dynamic testing of components of the RCPB to ensure it can withstand the applicable design-basis seismic and dynamic loads in combination with other environmental and natural phenomena loads without leakage, rapidly propagating failure, or gross rupture.

The methods of test and analysis employed to ensure the operability of mechanical and electrical equipment are based on joint standard International Electrotechnical Commission (IEC)/Institute of Electrical and Electronics Engineers (IEEE) 60980-344 (Reference 3-58). Regulatory Guide (RG) 1.100 endorses IEEE-344-2013. The BWRX-300 design utilizes IEC/IEEE 60980-344. The additional guidance provided in RG 1.100 is used to identify individual components of the RCPB demonstrating through testing and analysis, or a combination of both, that a given component will not leak as a result of any combination of loadings for which it is qualified. This ensures that components are tested to the highest quality standards practical.

Seismic design and design of Seismic Category I SSCs are addressed in NEDC-34165P Attachment 1 (Reference 3-1).

3.9.2.1 Seismic and Dynamic Qualification Criteria

A determination of the criteria for seismic and dynamic qualification is dependent on the type of equipment to qualify either mechanical, electrical, and/or I&C and the required seismic and dynamic inputs necessary to demonstrate structural and/or functional integrity. The criteria is provided in the following sections.

3.9.2.1.1 Qualification Standards

The guidance provided in the ASME BPVC Section III Division 1 "Rules for Construction of Nuclear Facility Components" is followed in the design of SC1 mechanical equipment to achieve the structural integrity of pressure boundary components. SC1 valves consider the qualification guidance provided in ASME QME-1, "Qualification of Active Mechanical Equipment Used in Nuclear Facilities," (Reference 3-59), for their qualification program.

Seismic and dynamic qualification of mechanical and electrical equipment and associated supports are considered for testing, analysis, or a combination of testing and analysis in accordance with IEC/IEEE 60980-344 (Reference 3-58) in accordance with RG 1.100.

Qualification by Actual Seismic Experience

IEC/IEEE 60980-344 (Reference 3-58) provides experience based seismic qualification methodology and is utilized as appropriate. In addition, ASME QME-1 (Reference 3-59) seismic experience may be utilised as appropriate. The information includes the credibility and completeness of compilation of the earthquake experience database for the seismic qualification of electrical equipment. The inclusion and exclusion rules for electrical equipment in the experience database, the justification used to demonstrate the similarity among the member items in a reference equipment class, the justification used to demonstrate the similarity between electrical equipment in the experience database and equipment in the NPP for seismic qualification purposes, and the justification used to demonstrate the functionality of equipment and the member items in a reference equipment class during and after a seismic event.

Qualification by Similarity

Qualification by similarity for Seismic Category I and 2 equipment is based on operating experience of similar equipment or to qualify multiple similar pieces of equipment by testing and/or analysing only one of the pieces of equipment. When extrapolation of data is made from similar equipment, a description of the differences between the equipment items involved is required. Justification that the differences do not degrade the environmental and/or seismic adequacy below acceptable limits and any additional supporting data is included.

Test results can be extrapolated for dynamic loading conditions in excess of, or different from, previous tests on a piece of equipment if the test results are in sufficient detail to allow an adequate dynamic model of the equipment to be generated. The model provides the capability of predicting failure under the increased or different dynamic load excitation. IEC/IEEE 60980-344 (Reference 3-58) defines the analytical method utilised in similarity qualification.

Functional Qualification of Active Mechanical Equipment

The seismic qualification of active mechanical equipment is performed considering the methods and requirements specified in ASME QME-1 (Reference 3-59).

3.9.2.1.2 Qualification Program

The equipment qualification program follows the requirements provided in IEC/IEEE 60780-323 (Reference 3-57), and is used to determine the overall equipment qualification test plan, including EQ provided in Subsection 3.9.3. The program meets the qualification criteria contained in IEC/IEEE 60980-344 (Reference 3-58) that includes seismic and dynamic mechanical and electrical equipment qualification.

3.9.2.1.3 Seismic Qualification Report

The seismic qualification report follows the requirements that are defined in IEC/IEEE 60980-344 (Reference 3-58) and is specific to the Seismic Category I electrical and mechanical equipment and associated supports to be qualified.

3.9.2.2 Methods and Procedures for Qualifying Mechanical and Electrical Equipment

3.9.2.2.1 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:

- The characteristics of the required input motion are dominated by one frequency
- The anticipated response of the equipment is adequately represented by one mode
- 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
- The time phasing of the inputs in the vertical or horizontal directions is 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 are used in determining Required Response Spectra (RRS) of input motion used for the test.

When both test and analysis are defined as acceptable methods, the deciding factors considered (as applicable) for choosing between tests or analysis includes:

- Magnitude of accelerations and frequency content of seismic and Reactor Building Vibration (RBV) dynamic loadings
- Environmental conditions associated with the dynamic loadings
- Nature of the function(s) required for a seismic event
- Size and complexity of the equipment
- Dynamic characteristics of expected failure modes (structural or functional)
- 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 analysed as an assembly). The replacement parts may be tested separately, if applicable.

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.

3.9.2.2.2 Qualification by Testing

Seismic qualification of mechanical and electrical equipment including I&C by testing is performed in accordance with the requirements of IEC/IEEE 60980-344 (Reference 3-58).

Interface Requirements

Intervening structures or components (such as interconnecting cables, bus ducts, conduits) that serve as interfaces between the equipment to be qualified and that are supplied by others, are not qualified as part of the seismic equipment qualification program. When applicable, accelerations and frequency content at locations of interfaces with interconnecting cables, bus ducts, and conduits are determined and documented. This information is specified in the form of interface criteria.

Test Methods

The test methods presented in IEC/IEEE 60980-344 (Reference 3-58) provide acceptable types of testing dependent on the type of motion selected based on the expected vibration environment and technical requirements of the specific application.

The preferred method for seismic testing is to use triaxial, multi-frequency testing. However, if justified, biaxial and single-axis testing is acceptable. If biaxial testing is justified to be used, then each test is performed in two steps, where the first step is to apply the input motion to both the vertical and horizontal axis simultaneously. For the second step, the test specimen is rotated 90 degrees in the horizontal plane, and a second test is performed with the input motion applied to the vertical and horizontal axis. Therefore, biaxial testing at a minimum requires twice the number of runs as triaxial testing. The preferred method for biaxial testing is independent, random tests.

For biaxial testing, when independent, random tests are not available, four tests are performed:

- With the inputs in phase
- With one input 180 degrees out of phase
- With the equipment rotated 90 degrees horizontally and the inputs in phase

- With the same orientation as in the step (3), but with one input 180 degrees out of phase

Selection of Test Specimen

Representative samples of equipment and supports are selected for use as test specimens. Variations in the configuration of the equipment are analysed with supporting test data. Test specimen assemblies that represent multiple configurations are configured to represent the "worst case" configuration. For example, these variations may include mass distributions that differ from one cabinet to another. Therefore, it is analysed and justified which mass distribution(s) results in the maximum stresses, such as response accelerations or frequency content, and this worst-case configuration(s) is used as the test specimen(s).

Mounting of Test Specimen

The test specimen is mounted to the test table so that the installed configuration, including interfaces, is adequately simulated and differences between the configuration are evaluated and resolved. If the test specimen is intended to be mounted to a panel or enclosure, the panel, enclosure, or a test fixture representative of the mounting conditions is included in the testing, unless justified. If the test specimen cannot be mounted directly to the table due to mounting constraints, an interposing test fixture is designed and used as the mounting interface. However, the equipment-to-fixture mounting condition is to simulate its installed configuration and cause no dynamic coupling to the equipment. If the equipment being analysed has no required orientation, the worst possible orientation is considered. The test specimen is considered to be in its operational configuration (i.e., filled with the appropriate fluid and/or solid). The investigation ensures that the point of maximum stress is considered. The test specimen mounting, and configuration includes hardware interface requirements. For interfaces that cannot be simulated on the test table, the accelerations and any resonances at such interface locations are recorded during the equipment test and documented in the test report.

Aging and Vibration Conditioning

The testing simulates the effects of aging. Equipment is reviewed in terms of design, function, materials, and environment for its specified application to identify potentially significant aging mechanisms.

If equipment is subjected to vibrational loads throughout its lifetime in its in-service mounted condition, then vibration aging to its end-of-life condition is performed prior to seismic qualification when required by the applicable qualification standard(s).

3.9.2.2.3 Qualification by Analysis

Qualification by analysis without testing may be acceptable on equipment that is only required to maintain its structural integrity to perform its safety function as described in IEC/IEEE 60980-344 (Reference 3-58).

Dynamic analysis or an equivalent static analysis is employed to qualify the equipment when analysis is chosen as the method for qualification. 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 RRS), the equipment is considered rigid. The search for the natural frequency is done analytically, if the equipment shape is 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 either dynamic or static analysis can be used, in general, the choice of the analysis is based on the expected design margin because the static coefficient method (the easiest to perform) is far 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 is 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 RRS at the equipment mounting location, at a conservative and justifiable value of damping.

Both types of analyses are to verify integrity of the equipment is maintained under low level earthquake loads, including appropriate RBV dynamic loads in combination with normal operating loads, and Safe Shutdown Earthquake (SSE) loads, including appropriate RBV dynamic loads, unless otherwise justified.

NEDC-34165P Attachment 1 (Reference 3-1) defines acceptable load combinations and methods for combining dynamic responses for mechanical equipment. The same criteria are acceptable for electrical equipment.

3.9.2.2.4 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 the results used 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 is used for qualification of larger electrical equipment support assemblies containing SC1 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. For this case, a test is run to determine if there are natural frequencies in the support equipment within the critical frequency range (any frequency below the cutoff frequency on the response spectrum). 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 SC1 equipment mounted in the assembly would operate without malfunctioning.

For digital I&C equipment qualification in a mild environment, analysis can be used in addition to testing if there is testing of an identical or similar item, or there is operating experience of equipment under identical or similar conditions with a supporting analysis to show that the equipment to be qualified is acceptable. Also, I&C equipment qualification can be performed using an analysis in combination with partial type test data that supports the analytical assumptions and conclusions.

3.9.2.3 Methods and Procedures of Analysis or Testing of Supports for Mechanical, Electrical Equipment and Instrumentation

Methods and procedures of analysis or testing of supports for mechanical and electrical equipment and instrumentation are in accordance with IEC/IEEE 60980-344 (Reference 3-58) and ASME QME-1 (Reference 3-59).

3.9.2.3.1 Supports for Battery Racks, Instrument Racks, Control Consoles, Cabinets, and Panels

SC1 control boards, panels, and racks should consider the qualification guidance provided in IEEE 420, "IEEE Standard for Design and Qualification of Class 1E Control Boards, Panels, and Racks Used in Nuclear Power Generating Stations," (Reference 3-60), for their qualification program.

3.9.2.3.2 Cable Trays and Conduit Supports

SC1 cables consider the qualification guidance provided in IEEE 383, "IEEE Standard for Qualifying Electric Cables and Splices for Nuclear Facilities," (Reference 3-61), and IEEE 384, "IEEE Standard Criteria for Independence of Class 1E Equipment and Circuits," (Reference 3-62), and test requirements in IEEE 1202, "IEEE Standard for Flame-Propagation Testing of Wire and Cable," (Reference 3-63) are used for the qualification program.

Supports provided by the equipment supplier to be used for the equipment is to be qualified in accordance with this section by the equipment supplier.

Seismic Category I supports (hangers) that support trays or conduit that carry safety circuits are designed and analysed to demonstrate qualification in accordance with IEEE 628, "IEEE Standard Criteria for the Design, Installation, and Qualification of Raceway Systems for Class 1E Circuits for Nuclear Power Generating Stations," (Reference 3-64).

SC1 connection assemblies consider the qualification guidance provided in IEEE 572, "IEEE Standard for Qualification of Class 1E Connection Assemblies for Nuclear Power Generating Stations and Other Nuclear Facilities," (Reference 3-65) for the qualification program as endorsed by RG 1.156, "Qualification of Connection Assemblies for Production and Utilization Facilities," (Reference 3-88).

3.9.2.3.3 Line Mounted Equipment

IEC/IEEE 60980-344 (Reference 3-58) identifies special consideration is required for line-mounted (piping and duct system) equipment regarding seismic qualification as the most critical seismic loading condition can occur as a result of the piping or duct system.

Guidance and further clarification for special considerations for line-mounted equipment are provided in IEC/IEEE 60980-344 (Reference 3-58) as well as IEEE 382, "IEEE Standard for Qualification of Safety-Related Actuators for Nuclear Power Generating Stations and Other Nuclear Facilities," (Reference 3-66). Line-mounted equipment may also include active mechanical equipment subjected to ASME QME-1 (Reference 3-59), including the QR-A Non-mandatory Appendix.

3.9.3 Environmental Qualification of Mechanical and Electrical Equipment

EQ includes the generation and maintenance of evidence to ensure SSCs can perform their intended design functions and remain fit for purpose in the conditions under which they are expected to perform. Subsection 3.9.2 provides the methodology and requirements used for the seismic and dynamic qualification of Seismic Category I Mechanical and Electrical equipment.

3.9.3.1 Mild Environment

An environment that would at no time be significantly more severe than the environment that would occur during normal plant operation, including AOOs, and does not give rise to significant aging mechanisms.

3.9.3.2 Harsh Environment

An environment that significantly changes from normal including design basis events and post-accident conditions as a result of a DBA.

3.9.3.3 AOO Environment

AOO environmental conditions are the service conditions as a result of an operational deviation expected to occur during the operating plant lifetime that do not lead to accident conditions.

The methodology and requirements apply to the EQ of SC1 mechanical and electrical equipment, including I&C, located in harsh and mild environments. IEC/IEEE 60780-323 (Reference 3-57), as endorsed by USNRC RG 1.89, “Environmental Qualification of Certain Electric Equipment Important to Safety for Nuclear Power Plants,” (Reference 3-69), defines the methodology and criteria used to qualify SC1 equipment for harsh and mild environments for the BWRX-300.

The environmental conditions in which the instrumentation and equipment of the SC1 systems operate are considered in establishing the component specifications. Instrumentation specifications are based on the worst expected ambient environmental conditions in which the instruments operate.

Qualification of mechanical equipment that performs a safety function is in accordance with ASME QME-1 (Reference 3-59).

3.9.3.4 Equipment Identification and Environmental Conditions

3.9.3.4.1 Equipment Identification

The equipment qualification program generates and maintains a list of SC1 equipment located in harsh and mild environments. The qualification plan includes the following parameters, at minimum and as applicable: the test and/or analysis sequence, environmental and/or seismic/dynamic or EMC requirements, test item functions, identification of industry codes and standards applicable to equipment, identification of the test equipment including description and calibration plan, and test item part numbers, quantity, mounting, and connection details.

3.9.3.4.2 Environmental Conditions

General Requirements

Environmental Design Bases

The environmental conditions consider normal, AOO, accident, and post-accident conditions, as applicable. Equipment located below the maximum flood level considers the effects of submergence and is qualified for flooding if it is required to function in this condition. Post-accident monitoring equipment considers the criteria for accident monitoring instrumentation and EQ guidance provided in IEEE 497, “IEEE Standard Criteria for Accident Monitoring

Instrumentation for Nuclear Power Generating Stations," (Reference 3-67), as endorsed by RG 1.97, "Criteria for Accident Monitoring Instrumentation for Nuclear Power Plants," (Reference 3-68).

The harsh environment qualification program verifies that the equipment is designed to be compatible and perform its safety functions during normal conditions, postulated environmental conditions, DBA, and post-accident conditions.

Equipment located within harsh environment conditions is exposed to environmental conditions including temperature, pressure, relative humidity, radiation, and chemical sprays.

Equipment determined to have a significant aging mechanism and located in a harsh environment account for the aging mechanism in the qualification program.

Aging mechanisms to be analysed for equipment located in a harsh environment include time-temperature degradation (thermal), cycle aging (wear), and normal radiation exposure.

Analysis is performed to identify the environmental design bases for AOOs, normal, accident, and post-accident environments as applicable.

Equipment is qualified to the worst-case environmental conditions for the areas in which they are located for the duration that they are required to perform their SC1 function.

The Safety Category 1 functions are either functional performance requirements or fail-safe requirements. A fail-safe SC1 function consists of not failing in a manner detrimental to plant safety, accident mitigation, or prevention of a SC1 function. The basis for the Safety Category 1 function is included in the qualification documentation.

Although EQ by testing or analysis is not required for SC2 and SC3 components, these components are designed for their expected duty cycle and environmental conditions over the design life of the plant with due consideration for maintenance and aging management. Additionally, SC2 and SC3 components that perform a SC1 function are qualified to the specified environmental conditions by testing or analysis.

The environments are considered for electrical and mechanical equipment in the EQ program such as temperature, pressure, humidity, chemical effects, radiation, and flooding effects. The EQ program uses the recommended environmental margins per IEC/IEEE 60780-323 (Reference 3-57), Table 1.

Aging requirements apply to SC1 equipment. For equipment located in harsh and mild environments, the effect of aging is be performed prior to DBA testing when a significant aging mechanism exists. Equipment is reviewed in terms of design, function, materials, and environment for its specified application to identify potentially significant aging mechanisms.

Equipment that could be exposed to radiation is environmentally qualified to a radiation dose that simulates the calculated radiation environment (normal and accident) that the equipment can withstand prior to completion of its required safety functions.

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. For equipment that is required to be functional post-accident, then the radiation dose is increased beyond the dose required for qualified life to envelop post-accident conditions as well, unless it is determined to cover post-accident conditions separately.

A mild radiation environment for electronic equipment is defined as a total integrated dose less than 10 gray (Gy) (1.0E03 rad), and a mild radiation environment for other equipment is less than 100 Gy (1.0E04 rad) as defined in RG 1.89 (Reference 3-69).

Electronic and electrical equipment are tested with the equipment energized and performing its safety function if the required total integrated dose exceeds the mild environment level.

This ensures equipment is qualified for the worst-case radiation with DBA margin per the requirements of IEC/IEEE 60780-323 (Reference 3-57).

Electromagnetic Interference / Radio Frequency Interference and Voltage Surges

EMC requirements apply to all levels of SC equipment, SC1, SC2, SC3, and SCN and provides qualification methods and implementation guidance. EMC qualifications for BWRX-300 design follow the requirements defined in (1) EPRI TR-102323, "Guidelines for Electromagnetic Compatibility Testing of Power Plant Equipment," (Reference 3-70), or (2) Military Standards MILSTD-461G, "Requirements for the Control of Electromagnetic Interference Characteristics of Subsystems and Equipment," (Reference 3-71), or (3) IEC-62003, "Nuclear Power Plants – Instrumentation, Control, and Electrical Power Systems – Requirements for Electromagnetic Compatibility Testing," (Reference 3-72). The qualification for Electromagnetic Interference/Radio Frequency Interference (EMI/RFI) and voltage surges for EQ equipment in harsh and mild environments is by test, consistent with USNRC RG 1.180, "Guidelines for Evaluating Electromagnetic and Radio-Frequency Interference in Safety-Related Instrumentation and Control Systems," (Reference 3-73) "Guidelines for Evaluating Electromagnetic and Radio Frequency Interference in Safety Related Instrumentation and Control Systems". EMC Qualification and Acceptance Testing includes tests for susceptibility and emissions. Susceptibility and emissions requirements are applied to all SC and SCN microprocessor-based I&C and electrical equipment.

3.9.3.5 Qualification Tests and Analyses

3.9.3.5.1 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 includes 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.

Tests are performed in accordance with applicable industry standards, such as IEC/IEEE 60780323 (Reference 3-57).

3.9.3.5.2 Qualification by Analysis

In general, analysis is used to supplement test data and the analytical techniques and modelling assumptions are, when possible, based on a correlation of the analytical approach with testing or operating experience performed on similar equipment or structures.

Seismic and dynamic qualification by analysis is described in Subsection 3.9.2.

For qualification by analysis, a logical assessment, or a valid mathematical model of the equipment to be qualified is required, and the basis for the analysis includes physical laws of nature, results of test data, operating experience, and condition indicators, as applicable.

Analysis of data and tests for material properties, equipment rating, and environmental tolerance are acceptable methods to be used to demonstrate qualification.

Analysis alone is not used to demonstrate the initial qualification for electrical equipment in a harsh environment.

3.9.3.5.3 Qualification by Operating Experience

Qualification by use of operating experience requires documented data to be available confirming to the following conditions are met:

- 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 are bounded by the product providing the operating experience
- The installed product in general, is removed from service and subjected to partial type testing to include the DBA environments for which the product is to be qualified

3.9.3.5.4 Combined Qualification

Combination of test and analysis is used when it is deemed practical to use both methods to complete the qualification. The combined qualification method can be used for qualification for larger electrical 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.

For digital I&C equipment qualification in a mild environment, analysis can be used in addition to testing if there is testing of an identical item of equipment under identical conditions or under similar conditions or operating experience with a supporting analysis to show that the equipment to be qualified is acceptable. Also, I&C equipment qualification can be performed using an analysis in combination with partial type test data that supports the analytical assumptions and conclusions.

3.9.3.5.5 Specific Equipment Requirements

Mechanical Equipment

SC1 mechanical equipment, which has the sole Safety Category 1 function of maintaining pressure integrity, and which is designed, fabricated, and qualified consistent with ASME BPVC, Section III, “Rules for Construction of Nuclear Facility Components,” (Reference 3-74), is considered qualified.

For mechanical equipment where the loading under normal service is more severe than loading under DBA, then the loading under normal service is considered in addition to the loading under DBA by test and/or analysis.

For mechanical equipment, the loading and capability under DBA conditions is analysed in the qualification process to establish the suitability of materials, parts, and equipment needed for safety 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 is qualified, non-metallic material capabilities, and the evaluation of environmental effects.

In addition, the qualification guidance provided in ASME QME-1 (Reference 3-59) is considered for qualification of SC1 valves and SC1 mechanical pipe supports. The qualification of non-metallic parts considers the qualification guidance provided in the Nonmandatory Appendix QR-B of ASME QME-1 (Reference 3-59).

Electrical Equipment

Additional qualification guidance is considered for specific electrical equipment, if applicable, as follows:

- RG 1.158 “Qualification of Safety-Related Vented Lead-Acid Storage Batteries for Nuclear Power Plants” (Reference 3-75), which endorses IEEE 535 “IEEE Standard for Qualification of Class 1E Vented Lead Acid Storage Batteries for Nuclear Power Generating Stations,” (Reference 3-76).
- RG 1.40 “Qualification of Continuous Duty Safety-Related Motors for Nuclear Power Plants,” (Reference 3-77), which endorses IEEE 334, “IEEE Standard for Qualifying Continuous Duty Class 1E Motors for Nuclear Power Generating Stations,” (Reference 3-78) if considered applicable to BWRX-300 design.
- RG 1.63 “Electric Penetration Assemblies in Containment Structures for Nuclear Power Plants,” (Reference 3-79), which endorses IEEE 317, “IEEE Standard for Electrical Penetration Assemblies in Containment Structures for Nuclear Power Generating Stations,” (Reference 3-80).
- RG 1.73 “Qualification Tests for Safety-Related Actuators in Nuclear Power Plants,” (Reference 3-81), which endorses IEEE 382, “IEEE Standard for Qualification of Safety-Related Actuators for Nuclear Power Generating Stations and Other Nuclear Facilities,” (Reference 3-82).
- RG 1.89 “Environmental Qualification of Certain Electric Equipment Important to Safety for Nuclear Power Plants” (Reference 3-69) which endorses IEC/IEEE-60780-323 (Reference 3-57) that includes IEEE 638 “IEEE Standard for Qualification of Class 1E Transformers for Nuclear Power Generating Stations,” (Reference 3-83).
- RG 1.213 “Qualification of Safety-Related Motor Control Centers for Nuclear Power Plants,” (Reference 3-84), considers conformance with the requirements of IEEE 649, “IEEE Standard for Qualifying Class 1E Motor Control Centers for Nuclear Power Generating Stations,” (Reference 3-85) if considered applicable to BWRX-300 design.
- RG 1.210 “Qualification of Safety-Related Battery Chargers and Inverters for Nuclear Power Plants,” (Reference 3-86), which endorses IEEE 650 “IEEE Standard for Qualification of Class 1E Static Battery Chargers, Inverters, and Uninterruptible Power Supply Systems for Nuclear Power Generating Stations,” (Reference 3-87).

Instrumentation and Control Equipment

Additional qualification guidance is considered for specific I&C equipment, if applicable, as follows:

- Control boards, panels, and racks classified as SC1 components – IEEE 420 (Reference 3-60) for their qualification program.

Qualification of computer based I&C systems is in accordance IEEE 7-4.3.2 “IEEE Standard Criteria for Programmable Digital Devices in Safety Systems of Nuclear Power Generating Stations,” (Reference 3-89). The EMC requirements are specified in RG 1.180 (Reference 3-73), IEEE 7-4.3.2 does not directly address RG 1.180 although the guidance in the RG is considered for I&C equipment.

When computer based I&C systems environmental type testing is performed:

- 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.

- The testing demonstrates performance of all safety function that affected by environmental factors under the environmental service conditions specified in the design specification. Software algorithms, which are tested during Verification and Validation (V&V) testing, are not tested unless their outputs exercise different hardware components which are affected impacted by environmental conditions.
- The testing exercises all portions of the system that are necessary to accomplish the safety functions and those portions whose operation or failure could impair the safety functions.
- The testing confirms the response of digital interfaces and verifies that the design accommodates the potential effect of environmental conditions on the overall response of the system.

When computer based I&C systems environmental type testing is performed, 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.

Cables, Raceways, Supports

For qualification of SC1 cables, the qualification guidance provided in IEEE 383 (Reference 3-61), and IEEE 384 (Reference 3-62) are considered. The test requirement guidance provided in IEEE 1202 (Reference 3-63) is used as a qualification program.

Seismic Category I supports (hangers) that support trays or conduit that carry SC1 circuits are designed and analysed to demonstrate qualification in accordance with IEEE (Reference 3-64).

Seismic Category II supports used for SCN raceway (conduit and cable tray) in Seismic Category I and II structures are analysed to withstand the effects of an SSE.

SC1 connection assemblies consider the qualification guidance provided in IEEE 572 (Reference 3-65) as endorsed by RG 1.156, “Qualification of Connection Assemblies for Production and Utilization Facilities”.

Line Mounted Equipment

Guidance in IEC/IEEE 60980-344, “IEEE/IEC International Standard-Nuclear Facilities Equipment Important to Safety-Seismic Qualification” (Reference 3-58) identifies that special consideration is required for line-mounted (pipe-supported) equipment regarding seismic qualification as the most critical seismic loading condition that occurs as a result of the piping or duct system. Guidance and further clarification for special considerations for line-mounted equipment is provided in IEEE 382 (Reference 3-66). Line mounted equipment also includes active mechanical equipment subjected to ASME QME-1 (Reference 3-59) including the Non-Mandatory Appendix QR-A.

3.9.4 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 representative interference levels applied for susceptibility testing.

Susceptibility testing allows assessment of equipment immunity to EMI/RFI and confirmation of its Surge Withstand Capability. Emissions testing provide assurance that equipment is compatible with the expected electromagnetic environment.

3.10 Inservice Monitoring, Tests, Maintenance, and Inspections

3.10.1 Safety Design Bases and Requirements

NEDC-34176P (Reference 3-13) provides the specific features of the inspections, tests, modelling, and monitoring programs for the BWRX-300 plant.

SSCs that have a shorter service lifetime 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, risk reduction, aging management, and easy-removable insulation for inspection, testing and maintenance.

In cases where SSCs are of SC and cannot be designed to support the desirable testing, inspection, or monitoring schedules, one of the following approaches shall be taken:

- Proven alternative methods, such as surveillance of reference items or use of verified and validated calculation methods, shall be specified
- Conservative safety margins shall be applied, or other appropriate precautions shall be taken, to compensate for possible unanticipated failures

3.10.2 Inservice Monitoring

The BWRX-300 levels of in-service monitoring for SSCs is related to the D-in-D DLs that are specified in Subsection 3.1.7 and associated classifications of SSCs in Subsection 3.2.2. Specifics on in-service monitoring are developed in the other PSR chapters. The design provides facilities for monitoring chemical conditions of fluids and of metallic and non-metallic materials.

3.10.3 Inservice Testing

In-service testing of certain ASME BPVC Section III, “Rules for Construction of Nuclear Facility Components,” (Reference 3-90) Division 1 pumps, valves, and snubbers (dynamic restraints) as applicable is performed in accordance with the ASME Operations and Management of Nuclear Power Plants (OM) code. In addition, in-service testing is performed in accordance with applicable 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 in-service testing to the extent practicable.

The in-service testing program includes periodic tests and inspections that demonstrate the operational readiness of certain SSCs 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 PSR chapters, but periodic and ISI and testing are established for:

- Nuclear pressure boundary components
- Containment components
- Containment structures
- Safety-related structures
- Balance-of-plant pressure boundary SC components or based on Aging Management requirements

3.10.4 Inservice Maintenance

Maintenance of the BWRX-300 Nuclear Power plant is based in part on the recommendations of the following publications:

- IAEA TECDOC-658, "Safety Related Maintenance in the Framework of the Reliability Centered Maintenance Concept," (Reference 3-91)
- IAEA Safety Standards Series, No. NS-G-2.6, "Maintenance, Surveillance and ISI in Nuclear Power Plants," (Reference 3-92)
- IAEA Safety Standards Series – GSR Part 2: "The Management System for Facilities and Activities," (Reference 3-93)

Baseline data will be gathered during initial testing and system commissioning of SSCs.

NEDC-34176P (Reference 3-13) provides programmatic requirements for in service maintenance.

3.10.5 Inservice Inspection

Mechanical components and equipment including heat exchangers, pipe supports, pumps, valves, and vessels, that are classified as ASME BPVC 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 and ASME Code OM prior to the component or equipment leaving the manufacturer's facility.

Non-Destructive Examination (NDE) methods are described within ASME BPVC-V 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.

NEDC-34176P (Reference 3-13) provides programmatic requirements for ISIs.

3.11 Compliance with National and International Standards

The specific PSR chapters provide prescriptive details that related to the BWRX-300 design features and their alignment with regulations including compliance with both national and international standards. PSR 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 BWRX-300 new nuclear plant.

3.11.1 Claims, Arguments, and Evidence Structure

3.11.1.1 Expectations

The "ONR Safety Assessment Principles (SAPs) for Nuclear Facilities," (Reference 3-94) identify ONR's expectation that a safety case should clearly set out the trail from safety claims, through arguments to evidence. This approach can be given as:

- **Claims** (assertions) are statements that indicate why a facility is safe
- **Arguments** (reasoning) explains the approaches to satisfying the claims
- **Evidence** (facts) supports and forms the basis (justification) of the arguments

3.11.1.2 Approach

GEH has structured its submission using the 'Claims, Arguments and Evidence', or CAE, approach that has been widely used in the licensing of recent nuclear power projects in the UK. The top-level claim, referred to as the Fundamental Objective, is provided below:

Fundamental Objective

The BWRX-300 is capable of being constructed, operated, and decommissioned in accordance with the standards of environmental, safety, security and safeguard protection required in the UK.

The Fundamental Objective is supported by the following Level 1 Claim for the PSR:

Level 1 Claim:

The safety risks to workers and the public during the construction, commissioning, operation and decommissioning of the BWRX-300 have been reduced as low as reasonably practicable (ALARP).

This is in turn supported by the following Level 2 Claims:

Level 2 Claims:

The functions of systems and structures have been derived and substantiated taking into account RGP and OPEX, and processes are in place to maintain these through-life (Engineering Analysis).

The BWRX-300 has been developed in accordance with approved procedures, with appropriate governance and assurance arrangements by a competent and clearly defined organisation (Safety Case Area).

A suitable and sufficient safety analysis has been undertaken which presents a comprehensive fault and hazard analysis that specifies the requirements on the safety measures and informs emergency arrangements (Safety Analysis).

Safety risks have been reduced as low as reasonably practicable.

These claims are then further subdivided and supported by arguments and evidence within the PSR chapters, although aspects of the evidence will only come available once the BWRX-300 enters the detailed design phase during site-specific licensing.

3.11.2 Numerical Targets

The ONR SAPs (Reference 3-94) introduce numerical targets that the ONR uses in assessing the acceptability of a facility or activity. These are provided in Appendix C.

It is the intention that in the next licensing phase, a set of numerical targets will be established that are based on the targets presented in the ONR SAPs. The general principle will be to establish targets equivalent to the Basic Safety Level provided by the SAPs along with the requirement that risks are ALARP. This is captured as a Forward Action Plan item shown in Appendix B.

3.11.3 Categorisation and Classification

The BWRX-300 approach to categorisation and classification has been described in Subsection 3.2. The ONR SAPs (Reference 3-94) set out UK regulatory expectations for categorisation and classification, with this being discussed in Appendix C.

It has been identified that there is no provision in the BWRX-300 approach to categorisation of safety functions to assign a normal operation safety function to anything other than Safety Category 3, other than in the case where the failure of the associated SSC has been demonstrated to be practically eliminated. A Forward Action Plan item has been raised to address this, as discussed further in Appendix C.

Table 3-1: 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 DECs to prevent core melt	Safety features for DECs without core damage (Safety Category 2)	Emergency operating procedures
Level 4b/DL4b	Control of DECs to prevent or mitigate the consequences of severe accidents	Safety features for DECs 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

Table 3-2: Safety Category for Functions Based on Defence Line Assignment

Safety Category	Defence Line 3 Functions	Defence Line 4a Functions	Defence Line 2/4b Functions	Normal Functions
1	Primary function Integral support functions			
2	• Post 72-hour primary and support functions	Primary function Integral support functions Post 72-hour primary and support functions		
3	• Post 7-day primary and support functions • Make-ready support functions	• Post 7-day primary and support functions • Make-ready support functions	Primary function Integral support functions Post 72-hour primary and support functions Post 7-day primary and support functions	• Normal functions that perform a fundamental safety function • Normal functions that maintain the reactor parameters
N			• Make-ready support functions	• Make-ready support functions

Table 3-3: 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 equivalent (3)	API 650 AWWA D100-11 ASME B96.1 or equivalent(3)	—	Manufacturer Specified Standards, e.g., ASME B31.1, AISC	—	—

Notes:

(1) Excluding the Steel-Plate Composite Containment Vessel (SCCV).

(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, “Design and Construction of Large, Welded, Low-Pressure Storage Tanks,” (Reference 3-50), API 650, “Welded Steel Tanks for Oil Storage,” (Reference 3-51), American Water Works Association (AWWA), “Welded Carbon Steel Tanks for Water Storage,” (Reference 3-52), and/or ASME B96.1 standards, “Welded Aluminum-Alloy Storage Tanks,” (Reference 3-53), as applicable.

(4) For Tubular Exchanger Manufacturers Association (TEMA)-style heat exchangers, both the ASME Code and TEMA standard, “Standards of the Tubular Exchanger Manufacturers Association,” (Reference 3-54) are considered. Other heat exchanger design styles/configurations are not subject to the TEMA standard.

(5) Acronyms used in Table 3-3 refer to the ASME BPVC “Section III – Rules for Constructions of Nuclear Facility Components,” (Reference 3-55) 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)

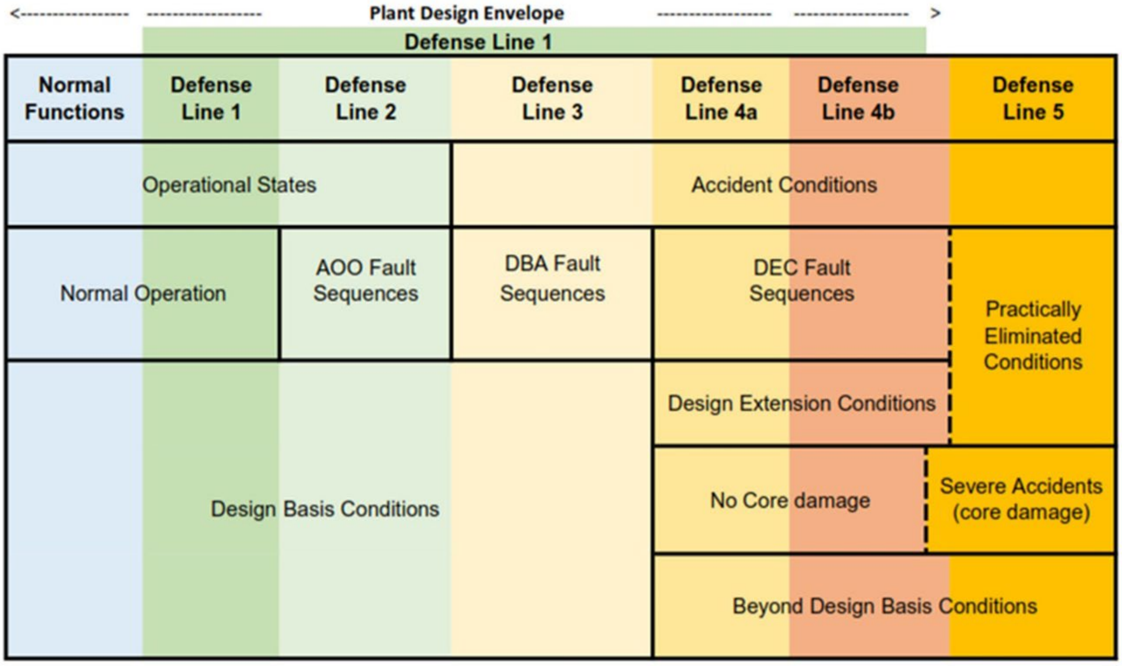

Figure 3-1: Defence-in-Depth - Plant States and Defence Lines

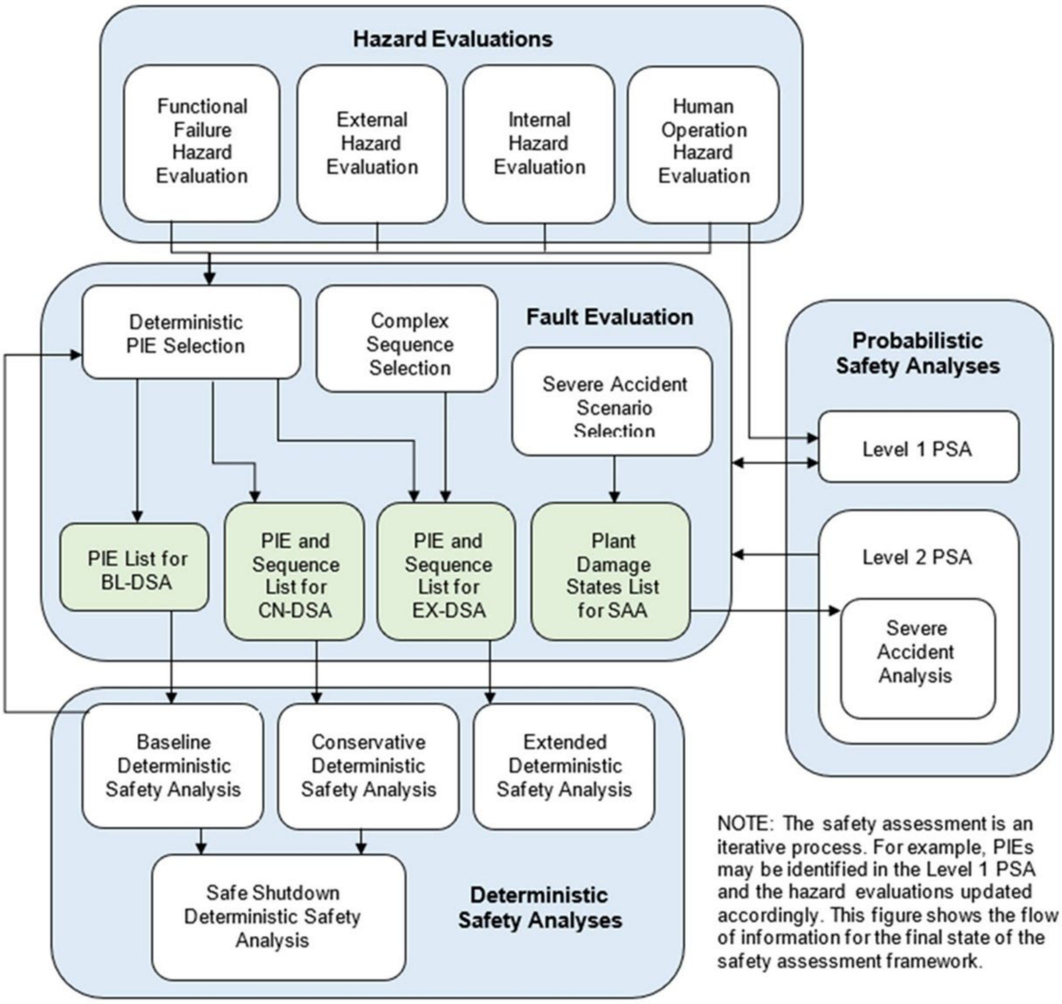

Figure 3-2: BWRX-300 Safety Strategy Implementation Process

3.12 References

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3-98 NEDC-34161P, Rev 0. "Comparison of BWRX-300 Approach to Categorisation & Classification with UK Expectations," September 2024.

3-99 NEDC-34357P, Rev 0. "BWRX-300 UK GDA Safety Case Manual Specification", April 2025.

3-100 005N9461 Rev 4. "BWRX-300 Structures, Systems, and Components Safety Classification," GE-Hitachi Nuclear Energy, Americas, LLC

Appendix A CLAIMS, ARGUMENTS AND EVIDENCE

A.1 Claims, Arguments, Evidence (CAE)

The ONR Safety Assessment Principles (SAPs) (Reference 3-94) identify ONR's expectation that a safety case should clearly set out the trail from safety claims, through arguments to evidence. The CAE approach can be explained as follows:

- Claims (assertions) are statements that indicate why a facility is safe
- Arguments (reasoning) explain the approaches to satisfying the claims
- Evidence (facts) supports and forms the basis (justification) of the arguments

The GDA CAE structure is defined within NEDC-34140P "BWRX-300 Safety Case Development Strategy," (SCDS) (Reference 3-95) and is a logical breakdown of the overall claim that:

> *"The BWRX-300 is capable of being constructed, operated and decommissioned in accordance with the standards of environmental, safety, security and safeguard protection required in the UK".*

This overall claim is broken down into Level 1 claims relating to environment, safety, security, and safeguards, which are then broken down again into Level 2 area related sub-claims and then finally into Level 3 (chapter level sub-claims).

The Level 3 sub-claims that this chapter demonstrates are identified within NEDC-34140P (Reference 3-95) and are as follows:

> *2.1.1: The safety functions (Design Basis) have been derived for the system/structure through a robust analysis, based upon RGP.*
>
> *2.1.3: The system/structure design has been undertaken in accordance with relevant design codes and standards (RGP) and design safety principles and taking account of OPEX to support reducing risks ALARP.*
>
> *2.1.4: System/structure performance will be validated by suitable testing throughout manufacturing, construction, and commissioning.*
>
> *2.1.5: Aging and degradation mechanisms will be identified and assessed in the design. Suitable examination, inspection, maintenance, and testing will be specified to maintain systems/structures fit-for-purpose through-life.*
>
> *2.4.1: RGP has been taken into account across all disciplines.*
>
> *2.4.2: OPEX and Learning from Experience (LfE) has been taken into account across all disciplines.*
>
> *2.4.3: Optioneering (all reasonably practicable measure have been implemented to reduce risk).*

In order to facilitate compliance, demonstration against the above Level 3 sub-claims, this PSR chapter has derived a suite of arguments that comprehensively explain how their applicable Level 3 sub-claims are met (see Appendix B).

It is not the intention to generate a comprehensive suite of evidence to support the derived arguments. However, where evidence sources are available, examples are provided.

A.2 Risk Reduction As Low As Reasonably Practicable

It is important to note that nuclear safety risks cannot be demonstrated to have been reduced ALARP within the scope of a two-step GDA. It is considered that the most that can be realistically achieved is to provide a reasoned justification that the BWRX-300 Small Modular

Reactor (SMR) design aspects will effectively contribute to the development of a future ALARP statement. In this respect, this chapter contributes to the overall future ALARP case by demonstrating that:

- The chapter-specific arguments derived may be supported by existing and future planned evidence sources covering the following topics:
  - RGP has demonstrably been followed
  - OPEX has been taken into account within the design process
  - All reasonably practicable options to reduce risk have been incorporated within the design
- It supports its applicable level 3 sub-claims, defined within NEDC-34140P (Reference 3-95).

Probabilistic safety aspects of the ALARP argument are addressed within NEDC-34178P (Reference 3-15).

Table A-1: Safety Objectives and Design Rules for SSCs Claims and Arguments

<table>
<tr><th>Level 3 Chapter Claim</th><th>Chapter 3 Argument</th><th>Sub-sections and/or reports that evidence the arguments</th></tr>
<tr><td colspan="3">2.1 The functions of systems and structures have been derived and substantiated taking into account RGP and OPEX, and processes are in place to maintain these through-life. (Engineering Analysis).</td></tr>
<tr><td>2.1.1: The safety functions (Design Basis) have been derived for the system/structure through a robust analysis, based upon RGP.</td><td rowspan="2">The BWRX-300 design has been assessed for development at Darlington, in Canada, and the Tennessee Valley, USA. It is designed based on US and Canadian nuclear regulatory requirements, along with international good practice. The UK BWRX-300 safety functions, and system/structure design are developed from these BWRX-300 principles with the consideration of UK context.</td><td>3.1 – General Safety Design Basis</td></tr>
<tr><td>2.1.3: The system/structure design has been undertaken in accordance with relevant design codes and standards (RGP) and design safety principles and taking account of OPEX to support reducing risks ALARP.</td><td>Chapter 3 – All sections.</td></tr>
<tr><td rowspan="2">2.14: System/structure performance will be validated by suitable testing throughout manufacturing, construction and commissioning.</td><td>The structural acceptance criteria for seismic category I structures have been considered and identified using good engineering practice guidance.</td><td>3.5 – Design of Seismic Category I Structures</td></tr>
<tr><td>The appropriate ASME Class has been considered in development of the test acceptance criteria for mechanical components.</td><td>3.6 – Mechanical Systems and Components</td></tr>
<tr><td>2.15: Aging and degradation mechanisms will be identified and assessed in the design.</td><td>Aging management topics to be covered as a minimum in design documents have been identified.</td><td>3.1.12 – Design Considerations for Aging Management</td></tr>
</table>

Level 3 Chapter Claim	Chapter 3 Argument	Sub-sections and/or reports that evidence the arguments
Suitable examination, inspection, maintenance and testing will be specified to maintain systems/structures fit-for-purpose through-life.	Aging and degradation considerations as part of equipment qualification by analysis or testing (or a combination) have been recognised.	3.9 – Equipment Qualification
	In-service examination, inspection and testing requirements will be developed taking cognisance of regulatory requirements and RGP.	3.10 – In-Service Monitoring, Tests, Maintenance, and Inspections.
	An effective maintenance, surveillance, inspection and testing; aging and degradation procedures can be developed to ensure the requirement of operating limits and conditions is effective.	3.1.10 – Design Approaches for the Reactor Core and for Fuel Storage. 3.1.12 – Design Considerations for Aging Management. Chapter 13, 13.3.2 – Maintenance, Surveillance, Inspection and Testing.
2.4: Safety risks have been reduced as low as reasonably practicable.		
2.4.1: RGP has been taken into account across all disciplines.	US and Canadian regulatory guidance, along with engineering good practice guidance, have been considered alongside the UK regulatory requirements.	Chapter 3
2.4.2: OPEX and LfE has been taken into account across all disciplines.	OPEX has been considered from decommissioning of existing facilities and incorporation of this ensured at the design phase to best facilitate the learning.	3.1.7 – Application of General Design Requirements and Technical Acceptance Criteria.
	The Safety Strategy principle for fuel handling and storage uses features proven through operating experience.	3.1.10 – Design Approaches for the Reactor Core and for Fuel Storage.
	The seismic equipment qualification methodology has considered and made use of actual seismic experience, using external seismic experience databases.	3.9.2 – Seismic and Dynamic Qualification of Mechanical and Electrical Equipment.

Level 3 Chapter Claim	Chapter 3 Argument	Sub-sections and/or reports that evidence the arguments
2.4.3: Optioneering (all reasonably practicable measures have been implemented to reduce risk).	RGP and guidance is considered as part of the selection process to ensure that the selected measure complies with the guidance.	Chapter 3 – All Sections

Appendix B FORWARD ACTIONS

FAP No.	Finding	Forward Action Plan Item	Delivery Phase
PSR3-1	Safety goals are currently set for the BWRX-300 target Core Damage Frequency and Large Release Frequency. Whilst these are useful metrics to assess, they do not allow comparison with the UK ONR SAP Numerical Targets 4-9 within the PSR.	Determine and justify the numerical targets to be adopted for the UK implementation of the BWRX-300 and document them in the specification for the safety case manual for implementation of the BWRX-300 in the UK. This action has been addressed during GDA Step 2 via production of NEDC-34357P “BWR-300 UK GDA Safety Case Manual Specification” Rev 0, which includes an outline of the approach to developing a radiological criterion suitable for defining the set of faults subject to design basis analysis and judging the effectiveness of the safety measures designated in the design basis analysis (as identified in FAP item PSR15.5-32 in PSR Subchapter 15.5). Note: detailed methods development and performance of analysis will be in a later licensing phase.	Completed within Step 2.
PSR3-448	The BWRX-300 design has been developed with reference to USNRC guidance rather than UK-specific guidance.	Justification is required as to why design against USNRC requirements and guidance is appropriate for UK deployment and that its use is in line with Regulatory Good Practice. Alternative codes and standards also need to be considered where appropriate to the UK.	Complete

Appendix C UK SPECIFIC CONTEXT INFORMATION

C.1 UK Context for Numerical Targets

C.1.1 ONR's Safety Assessment Principles and Numerical Targets

ONR's SAPs (Reference 3-94) introduce the numerical targets that ONR itself uses in assessing the acceptability of a facility or activity.

SAP NT.1 states:

> *Safety cases should be assessed against the SAPs numerical targets for normal operational, design basis fault and radiological accident risks to people on and off the site.*

ONR states in the SAPs that the targets should be used by inspectors *"… as an aid to judgement when considering whether radiological hazards are being adequately controlled and risks reduced to ALARP".*

Adding (para. 695):

> *The targets quantify ONR's risk policy and have been set to assist us in making proportionate regulatory decisions and targeting our resources to where the risks and hazards are greatest. More specifically, the targets are guides to inspectors to indicate where additional safety measures may need to be considered and, in the case of permissioning decisions, to help judge whether risks are tolerable.*

Normal operation – any person on the site	**Target 1**
The targets and a legal limit for effective dose in a calendar year for any person on the site from sources of ionising radiation are: Employees working with ionising radiation: BSL(LL): 20 mSv BSO: 1 mSv Other employees on the site: BSL: 2 mSv BSO: 0.1 mSv *Note that there are other legal limits on doses for specific groups of people, tissues and parts of the body (IRR17). Normal operational doses should also be reduced ALARP.*	

Normal operation – any group on the site	**Target 2**
The targets for average effective dose in a calendar year to defined groups of employees working with ionising radiation are: BSL: 10 mSv BSO: 0.5 mSv	

Normal operation – any person off the site	**Target 3**
The target and a legal limit for effective dose in a calendar year for any person off the site from sources of ionising radiation originating on the site are: BSL(LL): 1 mSv BSO: 0.02 mSv *Note that there are other legal limits to tissues and parts of the body (IRR17).*	

Design basis fault sequences – any person	**Target 4**
The targets for the effective dose received by any person arising from a design basis fault sequence are: **On site:** BSL: 20 mSv for initiating fault frequencies exceeding 1×10^{-3} pa 200 mSv for initiating fault frequencies between 1×10^{-3} and 1×10^{-4} pa 500 mSv for initiating fault frequencies between 1×10^{-4} and 1×10^{-5} pa BSO: 0.1 mSv **Off site:** BSL: 1 mSv for initiating fault frequencies exceeding 1×10^{-3} pa 10 mSv for initiating fault frequencies between 1×10^{-3} and 1×10^{-4} pa 100 mSv for initiating fault frequencies between 1×10^{-4} and 1×10^{-5} pa BSO: 0.01 mSv	

Individual risk of death from accidents – any person on the site	**Target 5**
The targets for the individual risk of death to a person on the site, from accidents at the site resulting in exposure to ionising radiation, are: BSL: 1×10^{-4} pa BSO: 1×10^{-6} pa	

Frequency dose targets for any single accident – any person on the site	**Target 6**
The targets for the predicted frequency of any single accident in the facility, which could give doses to a person on the site, are:	

Effective dose, mSv	**Predicted frequency per annum**	
	BSL	**BSO**
2–20	1×10^{-1}	1×10^{-3}
20–200	1×10^{-2}	1×10^{-4}
200–2000	1×10^{-3}	1×10^{-5}
> 2000	1×10^{-4}	1×10^{-6}

Individual risk to people off the site from accidents	**Target 7**
The targets for the individual risk of death to a person off the site, from accidents at the site resulting in exposure to ionising radiation, are: BSL: 1×10^{-4} pa BSO: 1×10^{-6} pa	

Frequency dose targets for accidents on an individual facility – any person off the site	**Target 8**
The targets for the total predicted frequencies of accidents on an individual facility, which could give doses to a person off the site are:	

Effective dose, mSv	**Total predicted frequency per annum**	
	BSL	**BSO**
0.1–1	1	1×10^{-2}
1–10	1×10^{-1}	1×10^{-3}
10–100	1×10^{-2}	1×10^{-4}
100–1000	1×10^{-3}	1×10^{-5}
>1000	1×10^{-4}	1×10^{-6}

Total risk of 100 or more fatalities	**Target 9**
The targets for the total risk of 100 or more fatalities, either immediate or eventual, from accidents at the site resulting in exposure to ionising radiation, are: BSL: 1×10^{-5} pa BSO: 1×10^{-7} pa	

ONR acknowledge in the SAPs that a safety case does not necessarily require detailed calculation for each target and that intermediate targets such as Core Damage Frequency (CDF) and Large Release Frequency (LRF) can be considered provided that *"…the overarching Principles EKP.1 to EKP.5 are not compromised through such approaches".*

As discussed in Subsection 3.1.4, the BWRX-300 has adopted stringent CDF and LRF targets of 1x10-6 per reactor-year and 1x10-7 per reactor-year respectively, along with the application of radiological protection principles to ensure that normal operational exposures reduced to levels that are ALARP.

C.1.2 Alignment of the BWRX-300 Safety Philosophy with ONRs Engineering Key Principles

The overarching Engineering Key Principles (EKPs) are listed below.

- EKP.1 (Inherent safety) - The underpinning safety aim for any nuclear facility should be an inherently safe design, consistent with the operational purposes of the facility
- EKP.2 (Fault tolerance) - The sensitivity of the facility to potential faults should be minimised
- EKP.3 (Defence in depth) - Nuclear facilities should be designed and operated so that defence in depth against potentially significant faults or failures is achieved by the provision of multiple independent barriers to fault progression.

- EKP.4 (Safety function) - The safety function(s) to be delivered within the facility should be identified by a structured analysis
- EKP.5 (Safety measures) - Safety measures should be identified to deliver the required safety function(s)

The overall safety philosophy for the design of the BWRX-300 is referred to as the Safety Strategy NEDC-33934P (Reference 3-39). The Safety Strategy ensures a consistent, robust, and systematic design approach and provides a framework for comprehensive and systematic safety assessments of the design. This is accomplished through the application of Safety and Design Principles based on the principles set forth in the IAEA document SSR-2/1 (Reference 3-40).

The BWRX-300 safety objective is to achieve a design with a very high level of safety with Safety and Design Principles based on a D-in-D approach consisting of five levels of defence called DLs. Safety is enhanced by deliberate design decisions informed by deterministic and probabilistic safety analyses, through an iterative safety framework wherein the design is implemented to meet defined safety objectives, which are confirmed via safety assessments. Results of safety assessments then provide feedback regarding the design and the process is repeated as required.

Design robustness is incorporated through appropriate design margins, and via DiD by the introduction of passive safety features which do not require dependence on external sources of power or operator actions to perform their stipulated functions.

C.1.3 Approach to Numerical Targets for the Preliminary Safety Report

There is a strong alignment between the BWRX-300 safety philosophy and the EKPs. This gives confidence that the application of stringent intermediate CDF and LRF targets, combined with the radiological protection principles and safety strategy for the BWRX-300 will ensure that Legal Limits are met and that risks can be demonstrated to be tolerable and ALARP.

The PSR, therefore, will continue to adopt these intermediate targets and provide a demonstration of risk in the context of CDF and LRF.

C.1.4 Approach to Numerical Targets for a Future Licensing Phase

It is the intention that in the next licensing phase (development of either a UK generic PCSR or site specific PCSR) a set of numerical targets will be established that are based on targets 1 to 9 presented in the SAPs. The general principle will be to establish targets equivalent to the Basic Safety Limit (BSL) combined with the requirement for the risks to be ALARP; the requirement to demonstrate an ALARP position is the overriding requirement, regardless of the position against the BSL or Basic Safety Objective (BSO).

This is addressed in NEDC-34357P “BWRX-300 UK GDA Safety Case Manual Specification” (Reference 3-99), which includes an outline of the approach to developing a radiological criterion suitable for defining the set of faults subject to design basis analysis and judging the effectiveness of the safety measures designated in the design basis (as identified in FAP item PSR15.5-32 in PSR Subchapter 15.5, Reference 3-20).

C.2 UK Context for ALARP

C.2.1 Legislative Basis for ALARP

The legislative basis of ALARP in the UK is derived from the “Health and Safety at Work etc. Act,” 1974 (Reference 3-96). The Act places duties on employers to ensure the health, safety, and welfare of their employees and to conduct their operations so that persons not in their employment are not exposed to risks to their health and safety. The employer is required to ensure that these duties are met So Far As Is Reasonably Practicable (SFAIRP), which is the basic legal requirement that each employer needs to conform to. In Office for Nuclear Regulation (ONR) guidance, the term ALARP is equivalent to SFAIRP.

C.2.2 ONR Safety Assessment Principles and ALARP

The ONR's Safety Assessment Principles (SAPs) for Nuclear Facilities (Reference 3-94), place the expectation that the safety case should provide an analysis of normal operation, potential faults and accidents, and of the engineering design and operations, and demonstrate the risks from all these perspectives have been reduced to ALARP.

The ALARP approach should include consideration of the following four aspects:

- Demonstration that international reactor OPEX has been taken into account in the overall design philosophy and in specific system designs
- Demonstration that RGP has been applied, including codes and standards comparison/justification
- Identification and evaluation of options (Optioneering)
- Risk assessment, as a way of understanding the significance of the issue to the holistic demonstration of ALARP i.e., to identify the severity of shortfalls against numerical targets, RGP, and/or deterministic rules

Following on from these is then the implementation of reasonably practicable improvements into the updated design reference.

In simple terms, the concept of ALARP is a requirement to take all measures to reduce risk where doing so is reasonably practicable. In most cases this is not done through an explicit comparison of costs and benefits, but rather by applying established RGP and standards. The development of RGP and standards includes ALARP considerations so in many cases meeting them is sufficient. In other cases, either where standards and RGP are less evident or not fully applicable, the onus is to implement measures to the point where the costs of any additional measures (in terms of money, time, or trouble – i.e., the sacrifice) would be grossly disproportionate to the further risk reduction that would be achieved (the safety benefit).

C.2.3 Approach to ALARP in the PSR

It is important to note that nuclear safety risks cannot be demonstrated to have been reduced ALARP within the scope of a PSR. It is considered that the most that can be realistically achieved is to provide a reasoned justification that the BWRX-300 SMR design aspects will effectively contribute to the development of a future ALARP demonstration.

The focus for the PSR, therefore, is to demonstrate that:

- operating experience has been taken into consideration
- the codes and standards used represent international Relevant Good Practice
- at a holistic level the fundamental design decisions made in the development of the BWRX-300 have the intent of reducing risks
- insights from the probabilistic risk analysis of previous generations of BWRs has been used to inform the BWRX-300 design

C.3 UK Context for Categorisation of Safety Functions and Classification of SSCs

C.3.1 ONR Safety Assessment Principles and Categorisation and Classification

UK regulatory expectations with respect to the categorisation of safety functions and classification of SSCs are set out in the Office for Nuclear Regulation's (ONR's) Safety Assessment Principles (SAPs) (Reference 3-94), with further guidance to inspectors provided in ONR Technical Assessment Guide 094 (TAG094), “Categorisation of Safety Functions and Classification of Structures, Systems and Components,” (Reference 3-97). TAG094 contextualizes the categorization of safety functions and classification of SSCs as a key activity in implementing a balanced approach to defence in depth in the design and operation of a nuclear facility, including Nuclear Power Plants (NPPs).

TAG094 sets out 5 high-level objectives for a scheme for categorization of safety functions and classification of SSCs:

- The systematic identification and categorisation of safety functions
- The systematic identification and classification of SSCs delivering those safety functions
- That the principle of D-in-D is applied, (with suitable and sufficient prevention, protection, and mitigation, in that order)
- That ALARP and RGP continue to always apply
- That classification informs the entire lifecycle of activities associated with SSCs

C.3.2 BWRX-300 Approach to Categorisation and Classification

The BWRX-300 schemes for the categorization of safety functions and the classification of SSCs are set out in Subsection 3.2.

The BWRX-300 approach to categorization of safety functions and classification of SSCs is based on the principles contained in IAEA SSR-2/1 (Reference 3-40) and IAEA SSG-30 (Reference 3-46). It can be summarized in three key steps:

- Functions that can impact nuclear safety are identified
- The identified functions are categorized (i.e., each assigned a safety category) based on their importance
- Safety classes are assigned to the components that perform the identified functions

In general, the approach provides a direct correlation between the DLs within which a safety function resides, which indicates its importance to safety, and its safety categorization, and then a direct linkage between the assigned safety categorization of a function and the classification of the SSC(s) through which it is delivered.

A specific review of the BWRX-300 approach to the categorization of safety functions and classification of SSCs against UK expectations has been performed and presented in NEDC-34161P, “Comparison of BWRX-300 Approach to Categorisation & Classification with UK Expectations,” (Reference 3-98).

This review shows that the BWRX-300 approach broadly aligns with UK expectations and meets the ONR’s 5 high-level objectives.

The review identified that there is a gap versus UK expectations with respect to normal operation safety functions:

- There is no provision in the BWRX-300 approach to categorization of safety functions to assign a normal operation safety function to anything other than Safety Category 3, other than in the case where the failure of the associated SSC has been demonstrated to be practically eliminated.

The potential impact of the identified gap has been considered and it is judged that based on more onerous reliability targets in BWRX-300 design compared with UK expectations and iterative confirmation of safety classifications as safety analysis progresses, there is confidence that the design of the BWRX-300 broadly aligns with UK expectations and will continue to do so.

There is judged to be no impact on the PSR, however closure of this gap will be required ahead of a future PCSR, and a Forward Action Plan item has been raised, as shown in NEDC-34161P (Reference 3-98).

GE Hitachi Nuclear Energy

NEDO-34165 (Attachment 1)
Revision C
July 2025

US Protective Marking: Non-Proprietary Information
UK Protective Marking: Not Protectively Marked

BWRX-300 UK Generic Design Assessment (GDA) Chapter 3 - Safety Objectives and Design Rules for Structures, Systems and Components (Attachment 1)

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EXECUTIVE SUMMARY

The purpose of Preliminary Safety Report (PSR) Chapter 3 is to present the general Safety Objectives and Design Rules for Structures, Systems and Components (SSCs) used in the design and assessment of the BWRX-300 reactor design.

This Attachment to Chapter 3 provides supplementary information in respect of:

- Protection Against External Hazards
- Protection Against Internal Hazards
- Design of Civil Structures
- Mechanical Systems and Components
- General Design Aspects for Instrumentation and Control (I&C) Systems and Components
- General Design Aspects for Electrical Systems and Components

The information in this Attachment provides topic specific information useful to a specialist reader whilst ensuring that Chapter 3 remains accessible to all readers.

ACRONYMS AND ABBREVIATIONS

Acronym	Explanation
AC	Alternating Current
ACI	American Concrete Institute
AISC	American Institute of Steel Construction
ALARP	As Low As Reasonably Practicable
ALWR	Advanced Light Water Reactor
ANS	American Nuclear Society
ANSI	American National Standards Institute
AOO	Anticipated Operational Occurrence
ARS	Acceleration Response Spectra
ASCE/SEI	American Society of Civil Engineers/Structural Engineering Institute
ASME	American Society of Mechanical Engineers
ASTM	ASTM International (formerly American Society for Testing and Materials)
ATH	Acceleration Time Histories
BDBA	Beyond Design Basis Accident
BPVC	(ASME) Boiler Pressure Vessel Code
BWR	Boiling Water Reactor
CAD	Computer Aided Design
CB	Control Building
CEUS	Central-Eastern US
CIV	Containment Isolation Valve
CLE	Checking Level Earthquake
CRD	Control Rod Drive
CRDH	Control Rod Drive Housing
CUF	Cumulative Usage Factor
CUW	Reactor Water Cleanup System
CWS	Circulating Water System
DBA	Design Basis Accident
DBE	Design Basis Earthquake
DBT	Design Basis Threat
DC	Direct Current
DCIS	Distributed Control and Information System
DEC	Design Extension Condition
D-in-D	Defence-in-Depth
DL	Defence Lines
DP-SC	Diaphragm Plate Steel-Plate Composition
ECCS	Emergency Core Cooling System

Acronym	Explanation
EDS	Electrical Distribution System
ELT	Emergency Level Transient
FE	Finite Element
FIRS	Foundation Input Response Spectra
FIV	Flow Induced Vibration
FLT	Faulted Level Transient
FMCRD	Fine Motion Control Rod Drive
FPS	Fire Protection System
FRMAC	Federal Monitoring and Assessment Center
GALE	Gaseous and Liquid Effluents
GDA	Generic Design Assessment
GEH	GE Hitachi Nuclear Energy
HCU	Hydraulic Control Unit
HVAC	Heating, Ventilation and Air Conditioning
I&C	Instrumentation and Control
IAEA	International Atomic Energy Agency
IBC	International Building Code
ICS	Isolation Condenser System
IEC	International Electrotechnical Commission
IEEE	Institute of Electrical and Electronics Engineers
ISI	In-Service Inspection
ISRS	In-Structure Response Spectra
IST	In-Service Testing
LB	Lower Bound
LMS	Lumped Mass Stick
LOCA	Loss of Coolant Accident
LS	Limit State
LV	Low Voltage
LWR	Light Water Reactor
MCR	Main Control Room
MSCIV	Main Steam Containment Isolation Valve
MV	Medium Voltage
NEI	Nuclear Energy Institute
NPP	Nuclear Power Plant
OBE	Operating Basis Earthquake
OM	(ASME) Operation and Maintenance of Nuclear Power Plants
ONR	(UK) Office for Nuclear Regulation
OPEX	Operational Experience

Acronym	Explanation
PBIRS	Performance Based Intermediate Response Spectra
PBSRS	Performance Based Surface Response Spectra
PER	Preliminary Environmental Report
PIE	Postulated Initiating Event
PMP	Probable Maximum Precipitation
PRA	Probabilistic Risk Assessment
PSA	Probabilistic Safety Assessment
PSR	Preliminary Safety Report
PWR	Pressurized Water Reactor
RAT	Reserve Auxiliary Transformer
RB	Reactor Building
RBV	Reactor Building Vibration
RCPB	Reactor Coolant Pressure Boundary
RCS	Reactor Coolant System
RDD	Radiological Dispersal Devices
RG	Regulatory Guide
RGP	Relevant Good Practice
RIV	Reactor Isolation Valve
RPV	Reactor Pressure Vessel
RWB	Radwaste Building
SAMG	Severe Accident Management Guidelines
SC1	Safety Class 1
SC2	Safety Class 2
SC3	Safety Class 3
SCN	Not Safety Classified
SC	Safety Classified
SCCV	Steel-Plate Composite Containment Vessel
SCR	Secondary Control Room
SDC	Shutdown Cooling System
SDG	Standby Diesel Generator
SIR	Seismic Interface Restraint
SMR	Small Modular Reactor
SRA	Site Response Analysis
SRP	Standard Review Plan
SRSS	Square-Root-of-the Sum of Squares
SSCs	Structures, Systems and Components
SSE	Safe Shutdown Earthquake
SSI	Soil-Structure Interaction

Acronym	Explanation
SSSI	Structure-Soil-Structure Interaction
TB	Turbine Building
TSV	Turbine Stop Valve
UAT	Unit Auxiliary Transformer
UB	Upper Bound
UK	United Kingdom
UPS	Uninterruptible Power Supply
U.S.	United States
USNRC	U.S. Nuclear Regulatory Commission
WUS	Western US
ZPA	Zero Period Acceleration

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REVISION SUMMARY

Revision #	Section Modified	Revision Summary
A	All	Initial Issuance
B	All	Update to reflect changes made across the PSR for end of GDA Step 2 consolidation.
C	All	Minor update to reflect minor typographical errors

3A SAFETY OBJECTIVES AND DESIGN RULES FOR STRUCTURES, SYSTEMS AND COMPONENTS – SUPPLEMENTARY INFORMATION

Introduction

The purpose of Preliminary Safety Report (PSR) Chapter 3 is to present the general Safety Objectives and Design Rules for Structures, Systems and Components (SSCs) used in the design and assessment of the BWRX-300 reactor design.

This Attachment to PSR Chapter 3 provides supplementary information in respect of:

- Protection Against External Hazards (Section 3A.3)
- Protection Against Internal Hazards (Section 3A.4)
- Design of Civil Structures (Section 3A.5)
- Mechanical Systems and Components (Section 3A.6)
- General Design Aspects for Instrumentation and Control Systems and Components (Section 3A.7)
- General Design Aspects for Electrical Systems and Components (Section 3A.8)

The main body of NEDO-34165, “BWRX-300 UK GDA PSR Chapter 3: Safety Objectives and Design Rules for SSCs,” (Reference 3A-1) follows the format provided in IAEA SSG-61, “Safety Standards – Format and Content of the Safety Analysis Report for Nuclear power Plants,” (Reference 3A-2). The information in this Attachment provides topic specific information useful to a specialist reader whilst ensuring that PSR Chapter 3 remains accessible to all readers. The section numbers in this Attachment align with those of PSR Chapter 3 to aid cross referencing.

This Attachment interfaces with the following PSR chapters:

- NEDC-34166P, “BWRX-300 UK GDA Chapter 4: Reactor (Fuel and Core),” (Reference 3A-3), NEDO-34167, “BWRX-300 UK GDA Chapter 5: Reactor Coolant System and Associated Systems,” (Reference 3A-4), NEDO-34168, “BWRX-300 UK GDA Chapter 6: Engineered Safety Systems,” (Reference 3A-5), NEDO-34169, “BWRX-300 UK GDA Chapter 7: Instrumentation and Control,” (Reference 3A-6), NEDO-34170, “BWRX-300 UK GDA Chapter 8: Electrical Power,” (Reference 3A-7), and NEDO-34171, “BWRX-300 UK GDA Chapter 9A: Auxiliary Systems,” (Reference 3A-8) - These chapters present the design of systems and components which are based on the relevant safety and design principles provided in PSR Chapter 3.
- NEDO-34172, “BWRX-300 UK GDA Chapter 9B: Civil Structures,” (Reference 3A-9) – presents the design of civil engineering works and structures which is based on the principles presented in this attachment.
- NEDO-34173, “BWRX-300 UK GDA Chapter 10: Steam and Power Conversion Systems,” (Reference 3A-10) – presents the design of the steam and power conversion systems which is based on the relevant safety and design principles presented in PSR Chapter 3.
- NEDO-34174, “BWRX-300 UK GDA Chapter 11: Management of Radioactive Waste,” (Reference 3A-11) – presents the design of systems and components containing radioactive materials based on the safety and design principles provided in PSR Chapter 3.

- NEDO-34175, "BWRX-300 UK GDA Chapter 12: Radiation Protection," (Reference 3A-12) – presents the design of radiation protection based on the safety and design principles presented in PSR Chapter 3.
- NEDO-34176, "BWRX-300 UK GDA Chapter 13: Conduct of Operations," (Reference 3A-13) – uses the relevant engineering substantiation principles presented in Chapter 3 to develop the operational conduct and management for the UK BWRX-300.
- NEDO-34177, "BWRX-300 UK GDA Chapter 14: Plant Construction and Commissioning," (Reference 3A-14) – presents the arrangements and requirements for plant construction and commissioning, considering the relevant principles presented in PSR Chapter 3.
- Chapter 15 – Safety Analysis (References 3A-15 through to 3A-24) – provides the overarching safety analysis including Probabilistic Safety Assessments (PSAs), Design Basis Analyses (DBAs), and Beyond Design Basis Accidents, including Design Extension Conditions and severe accidents, with the consideration of relevant principles presented in PSR Chapter 3.
- NEDO-34188, "BWRX-300 UK GDA Chapter 16: Operational Limits and Conditions," (Reference 3A-25) – uses the relevant engineering substantiation principles presented in PSR Chapter 3 to develop the operational limits and conditions for the UK BWRX-300.
- NEDO-34189, "BWRX-300 UK GDA Chapter 17: Management for Safety and Quality Assurance," (Reference 3A-26) – presents codes and standards applied to Management of Safety and Quality Assurance which is based on the selection principles of codes and standards in PSR Chapter 3.
- NEDO-34190, "BWRX-300 UK GDA Chapter 18: Human Factors Engineering," (Reference 3A-27) – presents the substantiation of Human Factors principles which are provided in PSR Chapter 3.
- NEDO-34191, "BWRX-300 UK GDA Chapter 19: Emergency Preparedness and Response," (Reference 3A-28) – presents the emergency preparedness and response required by the principles presented in PSR Chapter 3.
- NEDO-34192, "BWRX-300 UK GDA Chapter 20: Environmental Aspects," (Reference 3A-29) – presents the environmental aspects with consideration of the relevant principles presented in PSR Chapter 3.
- NEDO-34193, "BWRX-300 UK GDA Chapter 21: Decommissioning and End of Life Aspects," (Reference 3A-30) – presents codes and guidelines applied in decommissioning and end of life aspects based on the selection principles of codes and standards provided in PSR Chapter 3.
- NEDO-34194, "BWRX-300 UK GDA Chapter 22: Structural Integrity," (Reference 3A-31) – demonstrates the structural integrity by applying design requirements based on the relevant principles presented in PSR Chapter 3.
- NEDO-34195, "BWRX-300 UK GDA Chapter 23: Reactor Chemistry," (Reference 3A-32) – presents codes and guidelines applied in chemistry based on the selection principles of codes and standards provided in PSR Chapter 3.
- NEDO-34196, "BWRX-300 UK GDA Chapter 24: Conventional Safety and Fire Safety," (Reference 3A-33) – presents the applicable codes and standards in conventional safety and fire safety which are compliant with the selection principles of codes and standards provided in PSR Chapter 3.

- NEDO-34197, "BWRX-300 UK GDA Chapter 25: Security," (Reference 3A-34) – describes the general approach to security as well as physical and cybersecurity with consideration of the principles presented in PSR Chapter 3.
- NEDO-34198, "BWRX-300 UK GDA Chapter 26: Interim Storage of Spent Fuel," (Reference 3A-35) – presents applicable codes and standards in interim storage of spent fuel which are based on the selection principles of codes and standards presented in PSR Chapter 3.
- NEDO-34199, "BWRX-300 UK GDA Chapter 27: ALARP Evaluation," (Reference 3A-36) – presents the ALARP evaluation to support and assess the achievement of the nuclear safety objective provided in PSR Chapter 3.
- NEDO-34200, "BWRX-300 UK GDA Chapter 28: Safeguards," (Reference 3A-37) – demonstrates understanding of safeguards requirements at the generic level and how they are accommodated in the standard plant design, with consideration of the principles presented in PSR Chapter 3.

3A.3 Protection Against External Hazards

3A.3.1 Seismic Design

The seismic design of the BWRX-300 SSCs complies with the General Design Criteria 2, "Design Bases for Protection Against Natural Phenomena" of 10 CFR 50 Appendix A, "General Design Criteria for Nuclear Power Plants" (Reference 3A-149) as stated in 10 CFR 50, Appendix S, "Earthquake Engineering Criteria for Nuclear Power Plants" (Reference 3A-150).

The BWRX-300 SSCs are assigned a seismic category that reflects the SSC's functional and performance requirements during or after a seismic event and affects the rules that apply to their design.

Table 5.2 in DBR-0066822, "BWRX-300 System Functional Requirements," (Reference 3A-40) provides the seismic categorisation of the BWRX-300 systems and components. The seismic categorisation of the BWRX-300 Power Block structures is presented in Table 3A-1.

Per Table 3A-1, the containment structure, the containment internal structures and the Reactor Building (RB) enveloping them are the only civil structures categorised as BWRX-300 Seismic Category 1A. The Radwaste Building (RWB) which processes and houses liquid, solid and gaseous highly radiological waste is categorised as BWRX-300 Seismic Category RW. This category aligns with Seismic Category RW-IIa per US Nuclear Regulatory Commission (USNRC) RG 1.143, "Design Guidance for Radioactive Waste Management Systems, Structures, and Components Installed in Light-Water-Cooled Nuclear Power Plants," (Reference 3A-41). The RWB is designed for ½ Design Basis Earthquake (DBE). The Control Building (CB), Turbine Building (TB), Service Building and Reactor Auxiliary Structures support or protect SSCs that are not credited in the safety analysis or that can perform their desired function regardless of structure performance. These structures also have the potential to interact with the RB, along with the RWB, due to their proximity to the RB. The CB, TB, Service Building and Reactor Auxiliary structures are therefore categorised as BWRX-300 Seismic Category 2 and are designed in accordance with the International Building Code (IBC) (Reference 3A-43) or local building code. The seismic interaction evaluations of the RWB, CB, TB, Service Building and Reactor Auxiliary Structures with the RB structure are performed following the BWRX-300 specific requirements in Section 6.2 of NEDC-33914-A, "BWRX-300 Advanced Civil Construction and Design Approach," (Reference 3A-42). The remaining Power Block structures consisting of detached prefabricated/modularised Equipment Structures are categorised as BWRX-300 Seismic Category Non-Seismic (NS) and are designed per the IBC, "International Building Code," (Reference 3A-43) or local building code.

Table 3A-1 summarises the seismic design basis for the Power Block structures based on their seismic categories. As indicated in Table 3A-1 the seismic design of BWRX-300 Seismic Category 1A structures considers Limit State LS-D response defined in Table 1-1 of ASCE / SEI 43, "Seismic Design Criteria for Structures, Systems, and Components in Nuclear Facilities," (Reference 3A-44), as essentially elastic response. As explained in RG 1.208, "A Performance Based Approach to Define the Site-Specific Earthquake Ground Motion," (Reference 3A-45), 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

Analyses performed to qualify the BWRX-300 Seismic Category 1A structures are discussed in Section 3A.3.1.3. Analyses performed to qualify the BWRX-300 Seismic Category 1A and 1B subsystems are discussed in Section 3A.3.1.4. The interaction of BWRX-300 Non-Seismic Category 1A and 1B SSCs with BWRX-300 Seismic Category 1A or 1B SSCs is addressed in Sections 3A.3.1.3 ("Interaction of Non-Seismic Category 1A Structures with Seismic Category 1A/1B SSCs" and 3A.3.1.4 and 3A.3.1.3.8 ("Interaction of Non-Seismic Category 1A or 1B Subsystems with Seismic Category 1A or 1B Subsystems"). Seismic instrumentation used to monitor earthquake events, and their effects are discussed in Section 3A.3.1.5.

In the following sections, reference to the integrated RB is inclusive of the RB, containment, and containment internals, whereas RB is used to refer to the reactor building structure outside of containment.

3A.3.1.1 Standard Plant Design and Site-Specific Design

3A.3.1.1.1 Standard Plant Design (Generic Design Response Spectra)

As described in Section 3A.5.1.3.1, the Baseline (BL) 1 Standard Plant design is based on representative ground and seismic parameters. Generic Design Response Spectra (GDRS) have been used to develop the Standard Plant design, as described in 008N1829, "Standard Plant BWRX-300 Integrated Reactor Building Seismic Analysis Report," (Reference 3A-46). Such seismic parameters were developed following the approach in NEDC-33914A (Reference 3A-42) to cover the wide range of realistic seismological and geotechnical conditions that exist at most US candidate sites, with a baseline seismic resistance design (0.25g to 0.5g PGA) that is greater than the typical EUR values of 0.25g for all UK site conditions. However, the applicability of these generic parameters to a particular site would need to be evaluated during the review of any future site-specific license application. Section 3A.3.1.2 includes details of the GDRS used as input in the seismic design of BWRX-300 Seismic Category 1A buildings of the Standard Plant.

3A.3.1.1.2 Site Specific Design (Design Basis Earthquake)

When a site, or sites, are selected in the UK, a DBE will be developed that is appropriate to the site(s) in question. BWRX-300 Seismic Category 1A or 1B SSCs required to remain functional during and after a seismic event are seismically qualified to withstand the effects of a DBE. A DBE will be developed based on the results of site investigations. Section 3A.3.1.2 presents a summary of the approach to defining and using a DBE for the seismic design of the BWRX-300 Seismic Category 1A and 1B SSCs.

3A.3.1.2 Seismic Design Parameters

3A.3.1.2.1 Design Ground Motion

The BWRX-300 design considers Operating-Basis Earthquake (OBE) and DBE design ground motions following the regulatory guidance of NUREG-0800, "Standard Review Plan for the

Review of Safety Analysis Reports for Nuclear Power Plants," (Reference 3A-47), Standard Review Plan (SRP) 3.7.1, "Seismic Design Parameters". It is recognised that NUREG-0800 refers to the DBE as the "Safe Shutdown Earthquake" (SSE), and DBE used here is equivalent to SSE.

See PSR Chapter 15.8 (Reference 3A-24) for further discussion of the OBE. OBE loads are considered for post-flooding condition and cyclic loading considerations for the containment metal components following the regulatory guidance of NUREG-0800 (Reference 3A-47), SRP 3.8.2, "Steel Containment." The use of OBE for plant shutdown is discussed in Section 3A.3.1.5.3 ("Comparison of Measured and Predicted Responses and Conformance with RG 1.166").

3A.3.1.2.2 Design Response Spectra

GDRS

The ground motion design spectra presented in this section define the amplitude and the frequency content of the design ground motion used for the seismic design of the BWRX-300.

NEDC-33914-A (Reference 3A-42) Section 7.2 provides three sets of GDRS at the 5% damping level defining the horizontal and vertical components of the input ground motion at firm, median and hard sites.

The GDRS accommodate a wide range of sites with low-frequency amplification (deep soft profiles) and high-frequency amplification (shallow soft and stiff profiles) as well as small and large magnitude contributing sources. The multiple GDRS also accommodate the differences in spectral shape between Western US (WUS) and Central-Eastern US (CEUS) reference rock motions. Figure 4-1 in 008N1829 (Reference 3A-46) shows the GDRS defining the horizontal and vertical components of the ground motion for standard design of BWRX-300 located at firm, median, and hard sites.

The horizontal GDRS are anchored at 0.3 g PGA. The PGA value adopted is based on a review of publicly available data for ground motions at the finished grade of existing nuclear power plants and other nuclear facilities. The PGA value of 0.3 g appropriately reflects an upper range in seismic hazards for CEUS sites and is representative of the overall average hazard for WUS sites.

However, a PGA of 0.3 g may not envelop the seismic hazard at several hard rock high frequency candidate sites, where the PGA is considerably higher than 0.3 g. Accordingly, the Hard GDRS, which is associated with rock sites capable of delivering a high-frequency content motion, is multiplied by a scaling factor of 1.67 resulting in a PGA of 0.5 g for both the horizontal and vertical GDRS.

NEDC-33914-A (Reference 3A-42), Section 7.3 provides profiles of subgrade dynamic properties for the generic sites that are compatible with strain levels consistent with the GDRS. The properties reflect saturated soil conditions for the soil located below the ground water table which is assumed at plant grade.

Eight sets of subgrade profiles are provided representing a wide range of possible site conditions at the candidate sites. Figure 4-2 and Figure 4-3 in 008N1829 (Reference 3A-46) present the shear-wave velocity and P-wave velocity profiles respectively. Figure 4-4 in the same report presents the subgrade damping ratio profiles which are used for both Shear-wave and P-wave damping.

DBE

The ground motion design spectra representative of specific sites in the UK once known will be developed based on the results of a Probabilistic Seismic Hazard Assessment, considering as-built subgrade conditions consisting of engineered backfill on top of the existing fill and in-situ residual soil, weathered rock and base rock. The DBE ground motion design spectra are developed in accordance with NUREG-0800 (Reference 3A-47), SRP 3.7.1 and USNRC RG 1.208 (Reference 3A-45), using the performance-based approach in Section 2 of ASCE/SEI 43 (Reference 3A-44) and the additional requirements provided in Section 5.2.2 of NEDC-33914-A (Reference 3A-42).

In accordance with the requirements of NEDC-33914-A (Reference 3A-42), Section 5.2.2, the horizontal and vertical spectra defining the amplitude and frequency content of the site-specific DBE ground motion consist of the following:

- Foundation Input Response Spectra (FIRS) defining the DBE ground motion at bottom of the integrated RB foundation
- Performance Based Surface Response Spectra (PBSRS) defining the DBE ground motion at the finished plant grade elevation
- Performance Based Intermediate Response Spectra (PBIRS) defining the DBE ground motion at intermediate embedment depth elevation established, following the guidelines in NEDC-33914-A (Reference 3A-42), 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 the PBIRS is to ensure the ground motions used as input for the SSI analyses of deeply embedded integrated RB are adequate throughout the depth of the embedment.

The vertical FIRS, PBSRS and PBIRS for the BWRX-300 at the site are developed by applying Vertical to Horizontal (V/H) ratios from NUREG/CR-6728, "Technical Basis for Revision of Regulatory Guidance on Design Ground Motions: Hazard- and Risk-Consistent Ground Motion Spectra Guidelines," (Reference 3A-48), to the horizontal spectra.

The Nuclear Energy Institute (NEI) checks are performed following the procedure described in Section 5.3.4 of NEDC-33914-A (Reference 3A-42) to ensure the ground motion used as input for the deterministic SSI analyses of the deeply embedded integrated RB at its foundation bottom elevation meets the regulatory guidance of USNRC DC/COL-ISG-017, “Ensuring Hazard-Consistent Seismic Input for Site Response and Soil Structure Interaction Analyses,” (Reference 3A-49), to be hazard consistent with the results of the probabilistic Site Response Analysis (SRA). The NEI check requires that the PBSRS and PBIRS be enveloped by the FIRS propagated upward through the three strain-compatible soil profiles used as input for the SSI analyses. When the enveloped Acceleration Response Spectra (ARS) of the propagated motion do not meet or exceed the PBSRS or PBIRS, the FIRS are augmented until the augmented input ground motion satisfies the NEI check.

A check will also be done to ensure the derived horizontal and vertical FIRS envelope the spectra from USNRC RG 1.60, “Design Response Spectra for Seismic Design of Nuclear Power Plants,” (Reference 3A-50), anchored at 0.1g, as required by Appendix S to 10 CFR Part 50 (Reference 3A-150).

Design Time Histories

Design ground motion Acceleration Time Histories (ATHs) used as input to the seismic SSI analyses of the integrated RB at the site are developed by spectral matching seed ground motion records to the ground motion design response spectra. Per the guidelines of NEDC-33914-A (Reference 3A-42), Section 5.2.3, five sets of three design motion ATHs, 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. This number of design ground motion ATHs exceeds the minimum of four specified in NUREG/CR-5347, “Recommendations for Resolution of Public Comments, Seismic Design Criteria,” (Reference 3A-51).

Design ATHs are developed by fitting recorded seed time histories to the 5% damped target design spectra. Developed ATHs meet the acceptance criteria for Option 2 in Section II.1.B of NUREG-0800 (Reference 3A-47), SRP 3.7.1, Appendix F of RG 1.208 (Reference 3A-45), and the additional guidance of Section 5.2.3 of NEDC-33914-A (Reference 3A-42).

Using seed time histories, time histories compatible to the representative design ground motion spectra are generated using the time domain spectral matching method, “An Improved Method for Nonstationary Spectral Matching,” (Reference 3A-52), where adjustment of initial time histories (seed motions) is made by adding wavelet functions to the initial acceleration time history in time domain. This adjustment is repeated until its spectrum becomes compatible with the target spectrum over the desired frequency range. Per recommendations of NEDC-33914-A (Reference 3A-42), 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.

Response spectra of the generated ATHs are computed and compared to the appropriate target response spectra. Cross-correlation coefficients, peak values, Arias Intensity, and Power Spectral Density function are then computed for the spectrally matched time histories.

It will be demonstrated for ATH plots that the selected acceleration, velocity, and displacements are compatible and do not result in displacement baseline drift in conformance with the regulatory guidance of NUREG-0800 (Reference 3A-47), Section 3.7.1, Section II.1.B.

3A.3.1.2.3 Percentage of Critical Damping Values

The integrated RB structure and its common foundation are primarily constructed using an advanced Steel-Plate Composite system, referred to as a Diaphragm Plate Steel-Plate Composite (DP-SC) modular system. Refer to PSR Chapter 9B (Reference 3A-9) for further information on the DP-SC modules.

OBE and DBE damping values assigned to the integrated RB structures and components are in accordance with USNRC RG 1.61, “Damping Values for Seismic Design of Nuclear Power Plants,” (Reference 3A-53), with Diaphragm Plate Steel-Plate Composite (DP-SC) damping values being based on the values provided for Steel-Plate Composite walls in Tables 1 and 2 of USNRC RG 1.61 (Reference 3A-53).

Damping properties assigned to soil materials in the SSI analysis model consider the stress-strain properties corresponding to the level of seismic input per the guidance of NUREG-0800 (Reference 3A-47), SRP 3.7.1. Stiffness and damping properties of subgrade materials compatible to the strains generated by design level earthquake event are developed. The strain-compatible damping of the subgrade materials is limited to 15% in accordance with the regulatory guidance of NUREG-0800 (Reference 3A-47), SRP 3.7.1, USNRC RG 1.208 (Reference 3A-45), Appendix E and the recommendations of ASCE/SEI (American Society of Civil Engineers/Structural Engineering Institute) 4-16, “Seismic Design of Safety-Related Nuclear Structures,” (Reference 3A-54).

Following the regulatory guidance of USNRC RG 1.61 (Reference 3A-53), Section C.1.2 lower OBE damping ratios are used for generating in-structure demands for qualification of equipment and systems. The higher DBE damping values can be used for development of seismic demands for structural design per ASCE/SEI 43 (Reference 3A-44), Section 3.3.3 and USNRC RG 1.61 (Reference 3A-53), Section C.1.2, respectively.

Damping values for subsystems including piping and equipment are obtained using the procedures described in Section 3A.3.1.4.5 (“Analysis Procedure for Damping”).

3A.3.1.2.4 Supporting Media for BWRX-300 Seismic Category 1A Structures

The integrated RB structures are built on a common mat foundation deeply embedded into the ground. For a description of the generic soil profiles used to generate the Standard Design, refer to NEDC-33914-A (Reference 3A-42), Section 7. For the foundation embedment depth, dimensions, and total structural height of BWRX-300 Seismic Category 1A structures, refer to 007N7334, "Power Block General Arrangement," (Reference 3A-55).

For eventual site-specific assessments, uncertainties related to the determination and variation of subgrade conditions are addressed in the SSI analysis by using three sets of subgrade profiles, representing Lower Bound (LB), best estimate and Upper Bound (UB) estimates of the subgrade properties. The design uses an envelope of results from the SSI analysis of best estimate, LB and UB subgrade profiles to account for the variation and uncertainty in subgrade properties in accordance with Section 5.3 of NEDC-33914-A (Reference 3A-42).

3A.3.1.3 Seismic System Analysis

In accordance with the requirements and guidelines of NEDC-33914-A (Reference 3A-42), Section 5.0, the seismic analyses of the BWRX-300 Seismic Category 1A integrated RB consider the following effects on its structural integrity, seismic response, and distribution of stress demands:

- Interaction of the deeply embedded RB structure with the surrounding subgrade or Soil-Structure Interaction (SSI) effects
- Variation in the soil and structural parameters
- Hydrodynamic loads (mass and stiffness)
- Interaction of the RB with surrounding subgrade and other Power Block structures or Structure-Soil-Structure Interaction (SSSI) effects

To adequately account for the SSI and SSSI effects per guidance of NEDC-33914-A (Reference 3A-42), Section 5.1, the one-step approach, as defined in Section 3.1.2 of ASCE/SEI 4 (Reference 3A-54), is implemented for the seismic design of the integrated RB structures. Seismic structural stress demands are obtained directly from the results of SSI analyses of combined models that include Three-Dimensional (3D) Finite Element (FE) representations of the integrated RB and the surrounding soil, rock, and Power Block structures. The surrounding subgrade is represented by layered half-space continuum with equivalent linear elastic stiffness properties and damping.

To address the SSSI effects, seismic demands for the design of the RB SSCs are obtained from the responses of SSI analyses of the integrated RB FE model combined with simplified FE models of the surrounding Power Block structures.

The methodology used to develop the 3D integrated RB FE model is described in Section 3A.3.1.3 ("Procedures Used for Analytical Modelling").

3A.3.1.3.1 Seismic Analysis Method

The BWRX-300 one-step seismic SSI analysis approach provides demands for the seismic design and qualification of SSCs for all frequencies of interest and adequately addresses the effects of SSSI for the integrated RB with adjacent structures and foundations.

The BWRX-300 seismic analysis approach follows the guidance of NEDC-33914-A (Reference 3A-42), Section 5.0.

The seismic SSI and SSSI analyses are performed using the sub-structuring method in ASCE/SEI 4 (Reference 3A-54), Section 5.4, and the ACS SASSI (a system for analyses of SSI, see APPENDIX D) computer program to calculate the seismic response of the RB structure. 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 integrated RB structures, is obtained in the frequency domain for a selected set of frequencies and then interpolated for other frequency points.

The linear elastic assumption, based on the superposition principal, allows seismic demands obtained from the SASSI analyses to be combined with design demands obtained from other structural analyses in seismic design load combinations.

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

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

To account for the primary non-linearity of subgrade materials, a set of profiles of strain compatible subgrade properties are used as input for the SSI analysis. These subgrade properties are compatible to the strains generated by DBE level earthquake and are developed based on the results of equivalent linear probabilistic SRA.

As described in Section 3A.3.1.2.2 (Seismic Design Parameters – Design Response Spectra – Design Time Histories), five sets of three input motion ATHs are used as input for the seismic analyses to mitigate the uncertainty in the computed responses due to the phasing of the time history frequency components.

Input ground motion ATHs are applied to the SSI 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 probabilistic 1D SRA approach. The requirement of USNRC D/COL-ISG-017 (Reference 3A-49) for the ground motion used as input for the deterministic SSI analysis being consistent with the motions used for the probabilistic SRA is met.

Uncertainties related to variations of the input SSI parameters and the wave propagation pattern through the site are addressed by results of sensitivity analyses performed, following the guidance in Section 5.3 of NEDC-33914-A (Reference 3A-42).

Design responses are obtained from the analysis of each of the three ground motion components for each of the five sets of ATHs using the methodology described in the Sections below. The seismic design demands are calculated as the average of the results obtained from the analyses of the five sets of ATHs. Results of the sensitivity analyses are included in the RB seismic design basis when significant exceedances are observed.

Frequencies of Analysis

The frequencies of analyses are determined following the guidelines of Section 5.3.2 of NEDC-33914-A (Reference 3A-42).

The frequencies of analysis are selected at sufficiently small frequency intervals. Transfer function amplitude results for responses at the key locations 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.

3A.3.1.3.2 Natural Frequencies and Response Loads

Seismic responses of the BWRX-300 RB at the site are characterised using results of SSI analyses for responses at key structural locations and stress responses of key structural members selected per the guidance provided in Section 5.3.1 of NEDC-33914-A (Reference 3A-42).

3A.3.1.3.3 Procedures Used for Analytical Modelling

The 3D FE model used in the SSI analysis of the integrated RB meets the regulatory guidance of NUREG-0800 (Reference 3A-47), SRPs 3.7.1 and 3.7.2 and USNRC RG 1.61 (Reference 3A-53). The model also meets the structural modelling requirements of ASCE/SEI 4 (Reference 3A-54), Section 3, and the one-step modelling requirements and guidelines of NEDC-33914-A (Reference 3A-42), Section 5.1.1.

The 3D FE model used in the seismic SSI analysis of the integrated RB consists of the integrated RB structure, the surrounding subgrade, the excavated volume of the subgrade materials replaced by the embedded portion of the RB structure and the near field backfill materials. In addition to the integrated RB, simplified models of the surrounding RWB, CB, TB, Service Building, Reactor Auxiliary Structures and their foundations are included in the model to capture the SSSI effects in the RB seismic design per the guidance of NEDC-33914-A (Reference 3A-42), Section 5.3.7.

The integrated RB structure is primarily constructed of DP-SC modules. The DP-SC 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 as described in SCCV and Reactor Building Structural Design Report (Reference 3A-57), Section 3.4.

Dynamic Finite Element Modelling of Integrated RB

The details of the integrated RB Dynamic FE model are provided in PSR Chapter 9B (Reference 3A-9), Section 9B.2.5.1.

In accordance with NUREG-0800 (Reference 3A-47), SRP 3.7.2, Section II.3.D and ASCE/SEI 4 (Reference 3A-54), Section 3.4.2, the integrated RB structural FE model represents masses expected to be present at the time of the earthquake. These include 50% of the specified design live loads, 25% of the specified roof design snow loads and the inertia associated with the hydrodynamic effects of the fluids contained in various pools inside the RB.

The hydrodynamic effects that consist of the impulsive and convective (or sloshing) components are considered in accordance with Section II.1.A of NUREG-0800 (Reference 3A-47), SRP 3.7.2 and following the provisions of Section 3.6.3 of ASCE/SEI 4 (Reference 3A-54), and Chapter 5 of American Concrete Institute (ACI) 350.3, “Code Requirements for Seismic Analysis and Design of Liquid-Containing Concrete Structures,” (Reference 3A-58). The hydrodynamic mass is included in the model by:

- Distributing the horizontal impulsive fluid mass over the pool walls that are perpendicular to the direction of motion in accordance with the guidelines in ACI 350.3 (Reference 3A-58)
- Lumping the entire vertical fluid mass on the pool slab 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 walls in accordance with Section 9.4 of ASCE/SEI 4 (Reference 3A-54).

Beam and shell elements are used to adequately represent the configuration of all main structural members in the integrated RB, with the DP-SC shear walls, slabs, and mat foundation being modelled using thick shell elements. The FE model includes gross discontinuities such as large openings and member eccentricity. Member eccentricity and offsets are accounted for by using rigid beam and shell elements or by appropriately adjusting the properties of the centreline modelled elements to account for the effect of large 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 relieve 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 mat foundation
- Determine whether separation between RB shaft wall and soils occurs under SSE loading

The evaluation of effects of conditions at the contact interfaces with surrounding subgrade on the RB seismic response is discussed in Section 3A.3.1.3.9 (“Effects of Parameter Variations on Responses”).

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 PSR Chapter 9B (Reference 3A-9). BE stiffness properties are assigned to the DP-SC structures in accordance with the regulatory guidance of NUREG-0800 (Reference 3A-47), SRP 3.7.2 and the requirements of ASCE/SEI 4 (Reference 3A-54), Section 3.3.2.

The effective stiffness for analysis of the thick shell elements representing the DP-SC members is developed following the approach in Section 5.5 of the Reactor Building Structural Design Report (Reference 3A-57). The stiffness calculations account for the expected state of stress and level of cracking for different loading conditions during normal operation and accident conditions.

The effects of variation of structural stiffness and damping properties is considered in the modelling of the integrated RB to ensure accuracy of the calculated seismic responses and seismic demands. Section 5.3.5 in NEDC-33914-A (Reference 3A-42) describes methods used and sensitivity analyses performed to evaluate the effects on 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 SSCs. In accordance with the requirements of ASCE\SEI 4 (Reference 3A-54), 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 Section 3A.3.1.3.1 (Frequencies of Analysis).

Dynamic Modelling 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 NUREG-0800 (Reference 3A-47), SRP 3.7.2, Section II.3.B as stated in NEDC-33914-A (Reference 3A-42) and ASCE/SEI 4 (Reference 3A-54), Section 3.7. They depend 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 Reactor Pressure Vessel (RPV), the dynamic properties of the RPV and its components are represented by a Lumped Mass Stick (LMS) model capable of capturing the significant modes of the RPV seismic response. 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 stabilisers.

3A.3.1.3.4 Soil-Structure Interaction

The SSI analyses of the BWRX-300 Seismic Category 1A integrated RB are performed per the methodology presented in Section 3A.3.1.3, for the range of subgrade conditions presented in Section 3A.3.1.2.4 (“Supporting Media for Seismic Category 1A Structures”). The standalone model used for SSI analyses does not consider the SSSI effects of the surrounding Power Block structures.

To address the SSSI effects of the surrounding Power Block structures on the deeply embedded RB, SSSI analyses are performed on combined models that assume the Power Block structures surrounding the RB are supported by near surface mat foundations.

The seismic responses from the SSI and SSSI analyses are enveloped to provide design seismic demands for the deeply embedded RB.

Sensitivity studies performed to evaluate the effects of parameter variations on responses are discussed in Section 3A.3.1.3.9 (“Effects of Parameter Variations on Responses”).

Section 3A.3.1.3.9 (“Soil Separation Effects”) presents the BWRX-300 approach to determine if the separation at soil-structure interfaces can have significant effect on the seismic response.

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 throughout the life of the plant and to address specifics related to its design and construction.

Implementation of DC/COL-ISG-017 Guidance

BWRX-300 approaches for meeting USNRC DC/COL-ISG-017 (Reference 3A-49) guidance and addressing current limitations in USNRC DC/COL-ISG-017 related to the seismic analysis of deeply embedded structures, as identified in NUREG/CR-7193, “Evaluations of NRC Seismic-Structural Regulations and Regulatory Guidance, and Simulation-Evaluation Tools for Applicability to Small Modular Reactors,” (Reference 3A-56), Section 1.5.8 are described in NEDC-33914-A (Reference 3A-42), Section 5.3.4.

The intent of USNRC DC/COL-ISG-017 (Reference 3A-49) 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 the 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

Section 3A.3.1.2.2 (“Design Response Spectra”), using the procedure described in Section 5.3.4.1 of NEDC-33914-A (Reference 3A-42).

The augmented and smoothed horizontal and vertical 5% damped spectra 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.

Coupling of Soil and Structures

The seismic SSSI of the RB with the adjacent RWB, CB, TB, Service Building, Reactor Auxiliary Structures and Equipment Structures is explicitly addressed in the seismic analysis and design as described in Section 3A.3.1.3.4 (“Soil-Structure Interaction”).

Coarse FE models representing the BE dynamic properties of the surrounding buildings and foundations are included in the combined FE model used for the seismic SSSI analysis. These models are sufficiently refined to capture the global modes of vibration of the RWB, CB, TB, Service Building, Reactor Auxiliary Structures and Equipment Structures with significant (> 20%) modal mass participations in the three orthogonal directions.

Section 3A.3.1.3.8 (“Interaction of Non-Seismic Category 1A Structures with Seismic Category 1A and 1B SSCs”) presents the approach for addressing the requirements related to the seismic interaction of the RB with the surrounding RWB, CB, TB, Service Building and Reactor Auxiliary Structures and foundations.

3A.3.1.3.5 Development of In-Structure Responses

In-Structure Response Spectra (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 accordance with NUREG-0800 (Reference 3A-47), SRP 3.7.3, Section II.

In-Structure Response Spectra

The development of the ISRS follows the guidance of NUREG-0800 (Reference 3A-47), SRP 3.7.2, Section II.5, USNRC RG 1.122, “Development of Floor Design Response Spectra for Seismic Design of Floor Supported Equipment or Components,” (Reference 3A-59), and ASCE/SEI 4 (Reference 3A-54), Section 6.2. Equipment Structure Interaction effects may be considered for the development of ISRS using the guidance in Section 5.3.6 of NEDC 33914-A (Reference 3A-42).

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.

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.

In accordance with the requirements of ASCE/SEI 4 (Reference 3A-54), Section 6.2.1.1(a) and (b), the ISRS are developed from the calculated nodal in-structure responses by:

- 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
- Combining the co-directional ARS results obtained from the analysis with the three orthogonal components of seismic input motion using the Square-Root-of-the Sum of the squares (SRSS) method.

The ISRS are developed at small frequency intervals to ensure they are sufficiently close to the peak response frequencies of the supporting structure. To meet the regulatory guidance of NUREG-0800 (Reference 3A-47), SRP 3.7.2, Section II.5, 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. Following the guidance of Section 5.3 of NEDC 33914-A (Reference 3A-42), the enveloping ISRS are increased as described in Section 3A.3.1.3.9 (Effects of Parameter Variations) to address variations of important parameters. In accordance with USNRC RG 1.122 (Reference 3A-59) and ASCE/SEI 4 (Reference 3A-54), Section 6.2.3, the peaks of the enveloping ARS are broadened by a minimum of +/-15% to address uncertainties related to the modelling 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 (Reference 3A-54), 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 in Section 3A.3.1.3.5 (“In-Structure Response Spectra”).

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 shifting is applied to time histories to address uncertainties related to the modelling of natural frequencies of supporting structure

Relative Displacement

Relative Displacement between different support points of subsystems with multiple or distributed supports are evaluated using the support displacement time histories.

The time history of the relative displacements corresponding to each SSI analysis is obtained by algebraic calculation of the difference in displacement time histories at the support locations. Directional combination of the support displacement time histories is carried out on a time-step-by-time-step basis. Maximum design relative displacements are calculated as the envelope of the maximum relative displacements obtained for each SSI analysis case.

3A.3.1.3.6 Three Components of Design Ground Motion

Earthquake motion is three-dimensional and seismic design considers the effects of the 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 as discussed in Section 3A.3.1.2.2 (“Design Time Histories”). 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 (Reference 3A-54), Section 4.2.2, and USNRC RG

1.92, “Combining Modal Responses and Spatial Components in Seismic Response Analysis,” (Reference 3A-60).

- The time histories of responses due to the three earthquake components are combined algebraically on the time-step-by-time-step approach
- The maximum co-directional responses can be combined using the 100-40-40 method
- The maximum responses due to the three earthquake components can be combined using SRSS method

The absolute sum method used in time domain is implemented (e.g., for calculations of seismic demands for foundation bearing pressure and stability evaluations) as an alternative to performing the algebraic sum method for all possible combinations of the input motion directions.

3A.3.1.3.7 Combination of Modal Responses

Modal combination is not utilised for the analysis of the BWRX-300 Seismic Category 1A structures. These structures are evaluated using time history analysis in the frequency domain in which the equations of motion are solved for the selected range of frequencies defined is described in Section 3A.3.1.3.1 (“Frequencies of Analysis”).

3A.3.1.3.8 Interaction of BWRX-300 Non-Seismic Category 1A Structures with BWRX-300 Seismic Category 1A and 1B SSCs

To meet the interaction guidance of NUREG-0800 (Reference 3A-47), SRP 3.7.2, Section II.8, evaluations are performed of the lateral load resisting system of the RWB, CB, TB, Service Building and Reactor Auxiliary Structures following the approach in NEDC-33914-A (Reference 3A-42), Section 6.2. These evaluations are based on seismic responses of RWB, CB, TB, Service Building and Reactor Auxiliary Structures obtained from the SSSI analyses that incorporate the dynamic response of the RB and surrounding Power Block structures. As described in Section 3A.3.1.3.4 (“Soil-Structure Interaction”), models used in the SSI analyses of the RB include FE representations of the surrounding RWB, CB, TB, Service Building, and Reactor Auxiliary Structures. The FE models of the RWB, CB, TB, Service Building and Reactor Auxiliary Structures 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 (Limit State LS-C per ASCE/SEI 43 (Reference 3A-44)) structural response to calculate DBE demands for the main lateral load resisting structural members following the guidance of NEDC-33914-A (Reference 3A-42), Section 6.2.

The stability of foundations supporting the RWB, CB, TB, Service Building and Reactor Auxiliary Structures is checked using demands calculated per Section 3A.3.1.3.11(“Determination of Seismic Overturning Moments and Sliding Forces, Structure to Soil Pressures and Frictional Forces for BWRX-300 Seismic Category 1A Structures”). No reductions are applied to seismic driving force demands used for the stability evaluations to account for inelastic responses of these structures.

The gaps between the RB and adjacent structures are evaluated per guidance in NEDC-33914A (Reference 3A-42), 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.

3A.3.1.3.9 Effects of Parameter Variations on Responses

Uncertainties in structural properties, damping values, subgrade properties, and variations of SSI parameters are accounted for in the analysis and design of BWRX-300 Seismic Category 1A structures in accordance with NUREG-0800 (Reference 3A-47), SRP 3.7.2, Section II.4, ASCE/SEI 4 (Reference 3A-54), Section 5.1 and following the guidance of NEDC-33914-A (Reference 3A-42), Section 5.3.

Evaluations of effects of parameter variations on floor responses are based on comparisons of key in-structure responses defined as discussed in Section 3A.3.1.3.2 ("Natural Frequencies and Response Loads"), obtained from sensitivity SSI and SSSI analyses as described below.

Effects of Variation of Structural Stiffness and Damping Properties

Effective structural stiffness and lower OBE damping properties are assigned to the integrated RB FE model as discussed in Section 3A.3.1.3.3 ("Procedures Used for Analytical Modelling"). Effective stiffness assigned to concrete members considers the level of stress in the concrete members due to the most critical seismic load combinations following the recommendations in Section 5.3.5 of NEDC-33914-A (Reference 3A-42).

To address the effects of structural stiffness variations, sensitivity SSI and SSSI analyses are performed on models representing lower structural stiffness properties corresponding to accident thermal and high intensity load conditions. Higher DBE damping properties are used for the analysis of the model with LB structural stiffness as discussed in Section 5.3.5 of NEDC-33914A (Reference 3A-42).

Per the guidelines of NEDC-33914-A (Reference 3A-42), Section 5.3.5, sensitivity analyses are performed for BE subgrade profile. The effects of structural stiffness variations are assessed by comparing key in-structure responses 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. If the comparisons show significant exceedances (> 10%) in the RB seismic response due to the structural stiffness variations, the results of these sensitivity analyses are included in the RB seismic design basis.

Excavation Support and Backfill Effects

Excavation support and backfill effects are addressed following the guidelines of NEDC-33914-A (Reference 3A-42), Section 5.3.8.

Sensitivity seismic SSI analyses are performed using BE properties of surrounding in-situ subgrade materials on an 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 used to represent the BE dynamic properties of the excavation support structure. Solid elements are used to represent BE, and the dynamic properties of concrete fill material. The geometry of the excavation support and the lean concrete are modelled based on the nominal dimensions obtained from excavation plan drawings. To address the uncertainties related to the modelling of friction at the RB shaft interfaces, the sensitivity SSI analyses are performed considering two representative conditions:

- Fully bonded conditions assuming no slippage between the RB shaft and surrounding materials
- No-friction conditions assuming no friction resistance of RB shaft exterior walls

Results of these sensitivity analyses for key in-structure responses are compared with the corresponding results of the design basis SSI analyses of FE model that excludes the excavation support and 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 following the guidelines of NEDC-33914-A (Reference 3A-42), Section 5.3.10.

The seismic design of the RB is based on the analysis of models that reflect fully saturated conditions for 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 SSI and SSSI analyses of:

- Fully saturated soil profile with BE soil dynamic properties representative of accidental flood groundwater level
- 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 are compared with the results of the design basis SSI and SSSI 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 SSSI analysis of the BWRX-300 RB addresses the uncertainties related to the inability of linear models used for the seismic SSI analysis to explicitly represent the separation between the soil and the structure following the guidance of ASCE/SEI 4 (Reference 3A-54), Section 5.1.9(b), and Section 5.3.9 of NEDC-33914-A (Reference 3A-42).

A sensitivity SSSI 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 SSSI 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 (see Section 9B.2.5.2 of PSR Chapter 9B (Reference 3A-9)), 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 discussed in Section 3A.3.1.3.2 ("Natural Frequencies and Response Loads"), and stress demands calculated from this sensitivity analysis are compared to the corresponding results of the SSSI 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 SSSI 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 Wave

The potential for non-vertically propagating seismic waves at the site is assessed following the guidelines in Section 5.3.3 of NEDC-33914-A (Reference 3A-42), based on the geological and seismological conditions of a specific site.

Non-Linear Sensitivity Evaluation

A justification for the assumption of linear behaviour is made, considering the subgrade material within which the RB is embedded and the frequency content of the ground motion at a specific site.

3A.3.1.3.10 Methods Used to Account for Torsional Effects

Considerations are given in the modelling 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 as discussed in Section 3A.3.1.3.3 ("Dynamic Finite Element Modelling of Integrated RB").

In accordance with the requirements of ASCE/SEI 4 (Reference 3A-54), 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
- Distributions of structural mass and stiffness that differ from those represented in the 3D FE model used for the seismic response analysis

Following the regulatory guidance of NUREG-0800 (Reference 3A-47), SRP 3.7.2, Section II.11, the accidental torsional moment demands are 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 walls may conservatively be increased by 5% to account for the accidental torsion.

3A.3.1.3.11 Determination of Seismic Overturning Moments and Sliding Forces, Structure to Soil Pressures and Frictional Forces for BWRX-300 Seismic Category 1A Structures

Consistent with the regulatory guidance of NUREG-0800 (Reference 3A-47), SRP 3.7.2, Section II.14, contact spring elements installed in the one-step SSI models at interfaces between the structure and the subgrade are used for calculation of dynamic bearing pressure demands, seismic driving forces and overturning moments on the BWRX-300 Seismic Category 1A common mat foundation. Uncertainties related to friction at the RB shaft interfaces are addressed as described in Section 3A.3.1.3.9 (Excavation Support and Backfill Effects).

As described in Section 3A.3.1.3.6 ("Three Components of Design Ground Motion"), time histories of the horizontal and vertical seismic forces in the three directions are calculated as the algebraic sum of the 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.

The seismic inertia forces and overturning moments for the foundation stability evaluations and seismic bearing pressure calculations are obtained from SSI models with higher SSE 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 Section 3A.3.1.2.2 ("Design Time Histories"). 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 PSR Chapter 9B, Section 9B.1.3.2 (Reference 3A-9), are met.

The seismic bearing pressure demands are also calculated in the SSI analyses. 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 Section 3A.3.1.2.2 ("Design Time Histories"). The dynamic bearing pressure demand under each foundation is defined as the average of the results obtained from the five sets of ATHs.

3A.3.1.4 Seismic Subsystem Analysis

This section focuses on the seismic qualification by analysis of the BWRX-300 Seismic Category 1A and 1B subsystems which consists of:

- Distribution systems, including piping and supports, cable trays and supports, Heating, Ventilation and Air Conditioning (HVAC) ductwork and supports, conduits and supports
- Equipment supports
- Intervening structural elements between distribution systems and equipment supports and the RB structure
- RPV and internals

Input motions for the qualification of these subsystems are usually in the form of in-structure response spectra or ATHs obtained from the integrated RB seismic SSI analysis discussed in Section 3A.3.1.3. Input motions in terms of ATH are generally used. For other methods used in the seismic and dynamic qualification of subsystems, refer to Section 3.9 in PSR Chapter 3 (Reference 3A-1).

3A.3.1.4.1 Seismic Analysis Methods

Seismic analysis of subsystems is performed using one of the following methods in accordance with NUREG-0800 (Reference 3A-47), SRP 3.7.2, Section II.1:

- Time history analysis
- Response spectrum analysis
- Static coefficient

Time History Analysis

Assuming velocity proportional damping, the dynamic equilibrium equations for a lumped mass, distributed stiffness system are expressed in matrix form as: T[M] {ü(t)} + [C] {ú(t)} + [K] {u(t)} = {P(t)}

Where:

{u(t)}	=	time dependent displacement of non-support points relative to the supports
{ú(t)}	=	time dependent velocity of non-support points relative to the supports
{ü(t)}	=	time dependent acceleration of non-support 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 is 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 generalised 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 generalised 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 generalise coordinates. The physical solution is then given by the eigen transformation once the generalised 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 generalised coordinates. For systems subjected to short duration, high frequency excitation, 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 the time step (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.

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 modelling uncertainties in both the primary and secondary systems in accordance with USNRC RG 1.122 (Reference 3A-59) and ASCE/SEI 4 (Reference 3A-54), 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.

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 in accordance with NUREG-0800 (Reference 3A-47), SRP 3.7.2, Section II.1.B. 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 in accordance with NUREG-0800 (Reference 3A-47), SRP 3.7.2, Section II.1.B. For example, if the equipment is simple enough such that it behaves essentially as a single degree-of-freedom model and the fundamental frequency 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.

Seismic Analysis of Piping

The time history and the response spectrum methods are utilised in the piping analysis as required. The procedure for multi-support excitation described in Section 3A.3.1.4.9 ("Multiple-Supported Equipment and Components with Distinct Inputs") 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.

Secondary stresses due to anchor displacements may be calculated based on total anchor displacements or on a mode-by-mode basis. As a conservative approach, the maximum displacement of each support point can be computed by either time history analysis or response spectrum analysis for the supporting RB Structures. The computed individual maximum support displacements are then imposed on the supported piping and equipment systems in the most limiting, or severe, directional combination for the piping and equipment design verification consideration.

The dynamic loads on pipe mounted pumps and valves are determined by dynamic analysis of the piping system including the pumps and valves in the analytical model. If a pump or valve has a fundamental frequency greater than the cutoff frequency described in Section 3A.3.1.4.4 ("Basis for Selection of Frequencies"), it may be modelled as a rigid body. The pump or valve fundamental frequency is calculated for the equipment alone when it is rigidly supported at the nozzle where it interfaces with the piping system.

Seismic Analysis of Equipment

The time history and response spectrum methods are also utilised 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 is used.

The relative displacement between supports is determined from the dynamic analysis of the structure. The relative support-point displacements are used for static analysis to determine the additional stresses due to support displacements. As discussed in Section 3A.3.1.4, the static coefficient method may be used in lieu of the required dynamic analysis for rigid equipment and frame type structures supported at a single point.

Seismic qualification of equipment by testing is discussed in Section 3.9 of PSR Chapter 3 (Reference 3A-1).

Seismic Analysis of the Reactor Pressure Vessel and Internals

To capture the interaction of the integrated RB structures with the RPV and its internals components, such as fuel, guide tubes, and Control Rod Drive (CRD) System housing, a LMS model representing the RPV dynamic properties is included in the integrated RB model as discussed in Section 3A.3.1.3.3 ("Dynamic Modelling of Subsystems, Components and Equipment") and Section 3A.3.1.4.3 ("Modelling of Reactor Pressure Vessel and Internals").

The dynamic responses required as input for the design verification of RPV and internals components are generated directly in the dynamic analysis of the RB. Seismic ATH for the design of the RPV and internals are applied to the RPV and internals models simultaneously in the three orthogonal directions. Alternatively, two analyses can be performed to calculate the horizontal and vertical responses of the RPV separately.

The evaluation of RPV internals components in the horizontal direction is performed using a twostep analysis approach, where seismic accelerations at RPV support locations are applied to more detailed FE models of the RPV and internals. The first step of the two-step analysis consists of obtaining ATHs developed as described in Section 3A.3.1.3.5 ("Development of In-Structure Responses") at the RPV/RB interface locations from the RB SSI analyses. 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 is followed in the second step analysis of the RPV and internals. In accordance with the requirements of ASCE/SEI 4 (Reference 3A-54), Section 6.3.2, time step of the input time histories is varied to address uncertainties related to the modelling of natural frequencies of the supporting structure and the SSI analysis methodology as stated in Section 3A.3.1.3.5 (In-Structure ATH). .

3A.3.1.4.2 Determination of Number of Earthquake Cycles

The determination of the number of earthquake cycles for subsystem analysis is in accordance with Section II.2 of NUREG-0800 (Reference 3A-47), SRP 3.7.3, "Seismic Subsystem Analysis Review Responsibilities".

3A.3.1.4.3 Procedures Used for Analytical Modelling

The mathematical models for BWRX-300 Seismic Category 1A and 1B components are developed to realistically reflect the dynamic characteristics of the components.

To do so, each component is discretized into a series of interconnected beam elements or FEs. 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 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 below in Section 3A.3.1.4.4 ("Basis for Selection of Frequencies").

The node point spacing is selected such that the maximum length (L) of the FE between any two node points, in the direction of the stress wave propagation, satisfies the condition

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

Where λ and ν are the wavelength and wave velocity, respectively. The frequency f, or period T, corresponds to the cutoff frequency of Section 3A.3.1.4.4 ("Basis for Selection Frequencies").

Modelling of Piping Systems

The continuous BWRX-300 Seismic Category 1A and 1B piping systems are modelled 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. Concentrated weights on the piping system, such as the valves, pumps, and motors, are modelled as lumped mass rigid systems if their fundamental frequencies are greater than the cutoff frequency in Section 3A.3.1.4.4 ("Basis for Selection Frequencies”). 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 Section 3A.3.1.4.4(“Basis for Selection Frequencies”) when calculated as a simply supported beam with uniformly distributed mass.

All pipe guides and snubbers are modelled to produce representative stiffness to reduce modelling uncertainties. Snubbers are modelled 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.

Modelling of Equipment

For dynamic analysis, BWRX-300 Seismic Category 1A and 1B equipment is represented by lumped mass systems which consist of discrete masses connected by weightless beam elements and/or by any other appropriate FE representation. The criteria used to lump the masses are:

- 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 Section 3A.3.1.4.4 (“Basis for Selection Frequencies”).
- Mass is lumped at any point where a significant concentrated weight is located.
- 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.
- 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.

Modelling 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 Section 3A.3.1.4.3 (“Procedures Use for Analytical Modelling”).

The mathematical model of the RPV and internals consists of a LMS model connected by linear elastic members and 3D FE 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. 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 in the annulus. 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 modelled 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 modelled as an equivalent torsional spring.

Due to the small clearances in the horizontal directions, the fuel assembly is adequately modelled 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 modelled using FE or LMS methods.

3A.3.1.4.4 Basis for 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 Section 3A.3.1.3.1("Frequencies of Analysis"). For seismic load, modes up to 100 Hz are included. For other dynamic analysis, the cutoff frequency is 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 following the regularity guidance of NUREG-0800 (Reference 3A-47), SRP 3.7.3, Section II.4. Moreover, in any case, the equipment is analysed 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.

3A.3.1.4.5 Analysis Procedure for Damping

RPV and Internals Damping

Damping values are in accordance with ASCE/SEI 43 (Reference 3A-44) and are consistent with USNRC RG 1.61, "Damping Values for Seismic Design of Nuclear Power Plants," (Reference 3A-53), with the CRD damping values exceeding the regulatory guidance for a welded steel construction. Damping of fuel elements is in accordance with RG 1.61 (Reference 3A-53). See Table 3A-3 for RPV and internals damping values.

The Rayleigh damping is used for FE analysis of the RPV and internals when using the direct integration method.

Damping for Piping, Equipment and Equipment Support

Damping coefficients used in the seismic analysis of BWRX-300 Seismic Category 1A or 1B piping, equipment, equipment supports and intermediate structures between subsystems are presented in Table 3A-4. These are consistent with USNRC RG 1.61 (Reference 3A-53).

3A.3.1.4.6 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 Section 3A.3.1.3.6 ("Three Components of Design Ground Motion").

3A.3.1.4.7 Combination of Modal Responses

Modal combination procedures are in accordance with NUREG-0800 (Reference 3A-47), SRP 3.7.2, Section II.7 and USNRC RG 1.92 (Reference 3A-60).

3A.3.1.4.8 Interaction of BWRX-300 Non-Seismic Category 1A Subsystems with BWRX-300 Seismic Category 1A and 1B SSCs

The methodology used for evaluation and design of BWRX-300 Non-Seismic Category 1A subsystems interacting with BWRX-300 Seismic Category 1A or 1B SSCs is consistent with NUREG-0800 (Reference 3A-47), SRP 3.7.3, Section II.8.

3A.3.1.4.9 Multiple-Support Equipment and Components with District Inputs

The BWRX-300 approach for analysing equipment and components supported at several points by either a single structure or two separate structures conform with the regulatory guidance of NUREG-0800 (Reference 3A-47), SRP 3.7.3, Section II.9.

The time history direct integration, time history modal superposition and response spectrum modal superposition methods discussed in Section 3A.3.1.4.1 ("Seismic Analysis Methods") can be used in multi-support excitation analysis. However, the mode superposition procedure described in Section 3A.3.1.4.1 ("Seismic Analysis Methods") for an applied load vector is replaced with the corresponding mode superposition procedure for multi-support excitation analysis.

When using the time history or response spectrum methods, the following methods are acceptable:

- The uniform support motion method, where uniform time histories corresponding to the envelopes of all individual ISRS at the various support locations are applied at each attachment point simultaneously
- The independent support motion method, where 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 inertial (primary) and static stresses (secondary) due to differential displacements are separated. The inertial forces are used for primary stress calculations. For closely spaced modes of the supporting structure, the response is calculated as discussed in Section 3A.3.1.4.7 ("Combination of Modal Responses"). The total secondary stress for triaxial excitation is computed as the SRSS of the resultant secondary stresses for each excitation direction. The American Society of Mechanical Engineers (ASME) Boiler Pressure Vessel Code (BPVC) Code Section III, "Rules for Construction of Nuclear Facility Components," (Reference 3A-61), requires that the secondary stresses must be combined with the primary stress.

Using the response spectrum method, the support points response spectra are generated from support point ATH with ±15 percent peak broadening to account for the RB support structure modelling uncertainties in accordance with USNRC RG 1.122 (Reference 3A-59) and ASCE/SEI 4 (Reference 3A-54), Section 6.2.3. For uncorrelated dynamic inputs, the use of the SRSS method to combine modal responses is appropriate since the modal excitation maximum responses do not occur simultaneously and therefore are statically uncorrelated. For situation of high modal density, i.e., when the significant modes are "closely spaced", the modal responses could be highly correlated and the SRSS superposition may yield unconservative responses. In this case, the modal responses are initially combined algebraically, then combined with the uncorrelated responses using the SRSS method.

3A.3.1.4.10 Use of Equivalent Vertical Static Factors

Equivalent vertical static factors are used when the requirements for the static coefficient method in Section 3A.3.1.4.1 ("Static Coefficient") are satisfied.

3A.3.1.4.11 Torsional Effects of Eccentric Masses

Torsional effects of eccentric masses are considered in the modelling of subsystems as discussed in Section 3A.3.1.4.3 ("Procedures for Analytical Modelling").

3A.3.1.4.12 BWRX-300 Seismic Category 1A Buried Piping, Conduits and Tunnels

The BWRX-300 generic design does not include Seismic Category 1A buried pipes, conduits, or tunnels.

3A.3.1.4.13 Methods for Seismic Analysis of BWRX-300 Seismic Category 1A Concrete Dams

The design does not include nor require the presence of a dam.

3A.3.1.4.14 Methods for Seismic Analysis of Above-Ground Tanks

The BWRX-300 design does not include BWRX-300 Seismic Category 1A above-ground tanks.

3A.3.1.4.15 Effect 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 Section 3A.3.1.4.9 ("Multiple-Supported Equipment and Components with Distinct Inputs"), 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 Section 3A.3.1.4.8 ("Interaction of BWRX-300 Non-Seismic Category 1A Structures with BWRX-300 Seismic Category 1A or 1B SSCs") for the methodology used to obtain differential displacements used in the evaluation of subsystems.

3A.3.1.5 Seismic Instrumentation

Instrumentation is provided so that the seismic response of Safety Classified (SC) SSCs can be evaluated promptly after an earthquake.

The BWRX-300 seismic instrumentation system is consistent with NUREG-0800 (Reference 3A-47), SRP 3.7.4, "Seismic Instrumentation," and USNRC RG 1.12, "Nuclear Power Plant Instrumentation for Earthquakes" (Reference 3A-62). The instrumentation used for the measurements is capable of recording the effects produced by the most severe earthquakes that have been historically reported for the unique site considered and surrounding area, with sufficient margin for the limited accuracy, quantity and period in which historical data has been accumulated.

Provisions for in-service testing, remote alarm, recording capabilities, dynamic range and sampling rate with a low and adjustable actuating level or trigger, and data processing are in accordance with USNRC RG 1.12 (Reference 3A-62), and USNRC RG 1.166, "Pre-Earthquake Planning, Shutdown, and Restart of a Nuclear Power Plant Following an Earthquake," (Reference 3A-63).

Acceleration sensors are installed such that occupational radiation exposures associated with their location, installation, and maintenance are maintained As Low As Reasonably Practicable (ALARP).

3A.3.1.5.1 Location and Description of Instrumentation

Free-Field Instrumentation

The location of the free-field instrumentation follows the guidance of USNRC RG 1.12 (Reference 3A-62), Section C.1.2 and include:

- At least two triaxial accelerometers installed outside of the structure-ground interaction influence of the Power Block, but as close as practicable to the RB following the guidance of USNRC RG 1.12 (Reference 3A-62), Section C.1.3.1.
- At least one free-field downhole accelerometer installed at the depth of bottom of the RB foundation, below the free-field accelerometer at finished grade level, because the deeply embedded RB is founded at a depth more than 12 m (40 ft) below finished grade elevation.

Structure and Component Instrumentation

In accordance with the requirements of Section C.1.2 of USNRC RG 1.12 (Reference 3A-62), triaxial accelerometers are installed at several locations inside the RB:

- At least one accelerometer at the top of the RB base slab
- One accelerometer on the containment internal structure close to the reactor vessel
- One accelerometer in the RB close to the top of the containment internal structure
- One accelerometer close to the top of the containment structure
- One accelerometer at the top of the operating floor elevation

Triaxial accelerometers are installed at three locations outside the RB deemed important in accordance with RG 1.12 (Reference 3A-62), Section C.1.2.

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 in accordance with USNRC RG 1.12 (Reference 3A-62), Section C.1.3.2.

Most of the instrumentation stations are configured to be accessible for maintenance during full power operation in compliance with the guidance of USNRC RG 1.12 (Reference 3A-62), Section C.4.3. For sensors installed in inaccessible areas, provisions for data recording, external remote alarm indicating actuation and remote Inservice Inspections (ISIs) are provided.

Recording and Playback Equipment

Recording and playback units are provided for multiple channel recording and playback of the triaxial accelerometer signals. Characteristics of the recording and playback equipment follow the guidelines in USNRC RG 1.12 (Reference 3A-62), Section C.4.

3A.3.1.5.2 Control Room Operator Notification

Activation of the seismic trigger causes an audible and visual annunciation in the two control rooms to alert the plant operator that a felt earthquake has occurred in accordance with USNRC RG 1.12 (Reference 3A-62), Section C.7.

For other control room design requirements, refer to PSR Chapter 6 (Reference 3A-5).

3A.3.1.5.3 Comparison of Measured and Predicted Responses and Conformance with RG 1.166

The BWRX-300 pre-earthquake planning, and post-earthquake actions are in accordance with USNRC RG 1.166 (Reference 3A-63), including clarifications, and American National Standards Institute (ANSI)/American Nuclear Society (ANS)-2.23, "Nuclear Power Plant Response to an Earthquake," (Reference 3A-64), endorsed by USNRC RG 1.166 (Reference 3A-63).

Following a seismic event, operator actions and operator walkdown inspections are performed to assess the severity of the earthquake following the provisions of ANSI/ANS-2.23 (Reference 3A-64), Section 6. The data from the seismic instrumentation, coupled with information obtained from a plant walkdown, is 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. The plant is shut down if the walkdown inspections discover damage to equipment that would affect the safe operation of the plant, or if the OBE exceedance criteria determined in accordance with Section 6.4.1 of ANSI/ANS-2.23 (Reference 3A-64), are met.

Following plant shutdown, post-shutdown inspections and tests are performed in accordance with Sections 7.1, 7.2, 8.1 and 8.2 of ANSI/ANS-2.23 (Reference 3A-64) to determine the physical condition of the plant and its readiness to resume operation. Long term evaluations are performed following the requirements and guidelines of ANSI/ANS-2.23 (Reference 3A-64), Section 9. After plant is restarted (or prior to restart if the earthquake caused significant damage to the plant), long-term evaluations are carried out for engineering assessments of plant structures and equipment using the actual event records to assure their long-term reliability.

3A.3.1.5.4 Design, Installation and In-Service Surveillance

The seismic instrumentation is expected to operate during all modes of plant operation including periods of plant shutdown. The design, selection, and location of the BWRX-300 seismic instrumentation includes consideration for long service life, ease and low frequency of maintenance and calibration.

The instrumentation is designed and installed following the guidance of USNRC RG 1.12 (Reference 3A-62), Sections C.4 and C.5.

Prior to installation, the operational reliability of the seismic monitoring instrumentation is demonstrated in accordance with USNRC RG. 1.12 (Reference 3A-62), Section C.4.7.

Maintenance, testing and administrative procedures for seismic instrumentation defined in accordance with USNRC RG 1.12 (Reference 3A-62), Section C.9 are documented before the first facility start-up, and updated as necessary following any modification to the system. Following the guidance of Section C.3 of USNRC RG 1.12 (Reference 3A-62), the seismic instrumentation system components are maintained and tested to ensure that a maximum number of instruments are kept in service during plant operation and shutdown. The maintenance program addresses technical, testing, and administrative procedures.

The guidance in Appendix A to RG 1.166 (Reference 3A-63) is followed if the instrumentation is discovered to have been out of service during an earthquake.

3A.3.2 Extreme Weather Conditions

This section presents the design basis weather conditions considered in the design of the BWRX-300 SSCs for the representative extreme meteorological hazards identified in PSR Chapter 2 (Reference 3A-38).

3A.3.2.1 Temperature and Humidity

The extreme temperatures and humidity levels specified in PSR Chapter 2 (Reference 3A-38) are considered in the BWRX-300 design. Conservative safety margins are considered in the evaluations and design of SSCs to ensure their availability and efficiency under extreme temperature and humidity conditions.

3A.3.2.2 Rain

Rain load is considered in the design of the BWRX-300 building structures.

The RB roof is designed to minimise or eliminate rain loading, taking guidance from US NRC RG 1.102, "Flood Protection for Nuclear Power Plants," (Reference 3A-65) regulatory position 3, considering rain intensity and duration (Probable Maximum Precipitation - PMP) values listed in PSR Chapter 2 (Reference 3A-38).

Design for rain loading on the RWB roof is performed considering PMP values specified in PSR Chapter 2 (Reference 3A-38).

The rain loading on power block structures will be as a minimum based on BS EN 1991-1-1. Further details on design against rain loading is provided in PSR Chapter 15.8 (Reference 3A-23).

3A.3.2.3 Snow and Ice

The RB structure is designed using ground snow loads for normal and extreme winter precipitation discussed in PSR Chapter 2 (Reference 3A-38). For the RB structure, ground snow loads are converted to roof snow loading in accordance with the methodology specified in ASCE/SEI 7, "Minimum Design Loads and Associated Criteria for Buildings and Other Structures," (Reference 3A-66).

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 PSR Chapter 9B (Reference 3A-9)), 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 PSR Chapter 2 (Reference 3A-38).

The snow and ice loading on power block structures will be as a minimum based on BS EN 1991-1-1.

Further detail on design against snow and ice loading is provided in PSR Chapter 15.8 (Reference 3A-23).

3A.3.2.4 Design Wind and Extreme Wind Loadings

Wind loads are considered in the design of the BWRX-300 building structures and components.

3A.3.2.4.1 Wind Loadings

The design wind speed, its recurrence interval, the speed variation with height, and the applicable gust factors are used in defining the input parameters for the structural design criteria appropriate to account for wind loadings. The procedures that are utilised to transform the design wind speed into an effective pressure applied to structures take into consideration the geometrical configuration and physical characteristics of the structures and the distribution of wind pressure on the structures.

Design Wind Speed and Recurrence Interval

The design wind speed and recurrence interval used for the design basis are listed in PSR Chapter 15.8 (Reference 3A-23). Site-specific values reference the Generic Site Envelope (GSE) and are listed in PSR Chapter 2 (Reference 3A-38).

Determination of Applied Forces

Design wind speeds for the BWRX-300 Seismic Category 1A structures are translated into structural loading considering the methodology specified in ASCE/SEI 7-16 (Reference 3A-66) and is further discussed in PSR Chapter 2 (Reference 3A-38). Appropriate load combinations (including wind) to be used in design are given in American National Standards Institute/American Institute of Steel Construction (ANSI/AISC) N690-18, "Specification for Safety Related- Steel Structures for Nuclear Facilities," (Reference 3A-67). Procedures used are in conformance with Section II of NUREG-0800, SRP 3.3.1. However, NUREG-0800 (Reference 3A-47), SRP 3.3.1 refers to the ASCE/SEI 7-05, "Minimum Design Loads for Buildings and Other Structures," (Reference 3A-68) version. The BWRX-300 design uses the ASCE/SEI 7-16 (Reference 3A-66) version. A comparison performed between these two ASCE/SEI versions concludes that the ASCE/SEI 7-16 (Reference 3A-66) version bounds the ASCE/SEI 7-05 (Reference 3A-68) design parameters.

3A.3.2.4.2 Extreme Wind

The design input extreme wind parameters applicable to the BWRX-300 Seismic Category 1A structures are considered in PSR Chapter 2 (Reference 3A-38). Civil structures supporting and protecting SSCs for storage and processing of radioactive gas, liquid, and solid materials are classified in accordance with Regulatory Position 5 of RG 1.143 (Reference 3A-41). RW class SSCs are designed to withstand the extreme wind demand of Table 2 of RG 1.143 (Reference 3A-41). The Control Building (CB), including the Main Control Room (MCR) and egress route from the MCR to the Secondary Control Room (SCR) in the Reactor Building (RB), is designed to maintain its structural integrity under an extreme wind event to prevent incapacitating injury to MCR occupants or their egress to SCR in the RB.

Determination of Forces on Structures

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, "Tornado and Extreme Wind Design Criteria for Nuclear Power Plants," (Reference 3A-69) and follow the guidance of RG 1.76, "Design-Basis Tornado and Tornado Missiles for Nuclear Power Plants," (Reference 3A-70).

The procedure for transforming the extreme wind-generated missile impact into an effective or equivalent static load on structures is given in Section 3A.3.2.7. The extreme wind missile spectrum is discussed in Section 3A.3.2.7.

The individual extreme wind loading components, and the load factors are in accordance with NUREG-0800 (Reference 3A-47), SRP 3.3.2/ BC-TOP-3-A (Reference 3A-69) and are selected to produce the most adverse total extreme wind effect on structures. The combined extreme wind effects are determined based on Equations 1 and 2 provided in NUREG-0800 (Reference 3A-47), SRP 3.3.2. The factor of 1.2 specified in BCTOP3-A (Reference 3A-69), Sections 3.4 and 3.5.2 is applied to recognise potential non-conservative results for external walls when using the formulas specified in Appendix D of BCTOP-3A (Reference 3A-69) due to potential three-dimensional effects. As the formulas in Appendix D of BCTOP-3A (Reference 3A-69) are not used the 1.2 factor is not applied for the BWRX300 design.

NUREG-0800 (Reference 3A-47), SRP 3.3.2 invokes ASCE/SEI 7-05 (Reference 3A-68) for calculating maximum velocity pressure, q_z. After the values for coefficients, factors, and importance factor are substituted into the equation from SRP 3.3.2, the equation for maximum velocity pressure becomes $q_z = 0.00256V^2$, where V is the maximum extreme wind speed.

This aligns with Equation 3-8 of BC-TOP-3-A (Reference 3A-69). The size coefficient, C_s, from BC-TOP-3-A considers the load distribution length and the radius of maximum extreme wind. This size coefficient is not explicitly specified in ASCE/SEI 7-05 (Reference 3A-68). However, the values of external pressure coefficient, C_p, in Section 4.3.2 of BC-TOP-3-A (Reference 3A-69), are similar to those provided for wall pressure coefficients in Figure 6-6 of ASCE/SEI 7-05 (Reference 3A-68). The approach in BC-TOP-3-A (Reference 3A-69) and ASCE 7-05 are similar. Therefore, BC-TOP-3-A (Reference 3A-69) is considered to be appropriate for use in the BWRX-300 design.

As discussed in Section 3A.3.2.4, the BWRX-300 design uses the ASCE/SEI 7-16 (Reference 3A-66) version, which bounds the ASCE/SEI 7-05 (Reference 3A-68) design parameters.

Extreme wind loading considers extreme wind pressures, differential pressure loads due to rapid atmospheric pressure change, and tornado or wind-generated missile impact. For the purpose of structural analysis, the BWRX-300 RB is an enclosed (unvented) structure. The exposed exterior roof and walls of the RB are designed for the full pressure drop as provided in PSR Chapter 2 (Reference 3A-38).

Effects of Failures of Structures or Components Not Designed for Extreme Wind Loads

The BWRX-300 Seismic Category 2 power block structures surrounding the RB are designed to maintain their structural integrity during an extreme wind event such that they do not collapse on the BWRX-300 Seismic Category 1A RB.

Wind interaction evaluation follows the interaction requirements and guidance per the methodology described in NEDC-33914-A (Reference 3A-42), “BWRX-300 Advanced Civil Construction and Design Approach”, Section 6.0.

3A.3.2.5 Lightning

Grounding and lightning protection systems are used to protect structures, transformers and equipment against lightning induced surges as described in PSR Chapter 8 (Reference 3A-7).

Protection measures against fires and electromagnetic compatibility issues that could affect the functionality of electrical systems as a result of lightning are addressed in Section 3A.3.2.8 and Section 3A.3.3.1.

3A.3.2.6 Extreme Hydrological Conditions

Potential sources of external floods considered in the BWRX-300 design are discussed in PSR Chapter 2 (Reference 3A-38).

The integrated RB structure is designed to withstand the maximum external flood and groundwater levels specified in PSR Chapter 2 (Reference 3A-38).

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

- Hydrostatic and hydrodynamic loads to design walls below flood level.
- No exterior access openings below grade.

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.

Analysis Procedures for Protection of Structures Against Flood from External Sources

The values for the maximum external flood and maximum groundwater level parameters are provided in PSR Chapter 2 (Reference 3A-38). The maximum flood level and local intense precipitation elevations do not exceed plant grade, so there are no external flooding protection features required in the BWRX-300 design, and no need for a permanent dewatering system. All exterior access openings and exterior penetrations for the Seismic Category RB are above grade.

The BWRX-300 RB, which is embedded partially below grade, is analysed, and designed to withstand the effects of a maximum external flood defined at grade. Since the flood level is considered at 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 BWRX-300 design considers the groundwater level at finished grade elevation. 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. 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.

3A.3.2.7 External Missiles

Missiles considered are those that could result from a plant-related failure or incident including failures within and outside of containment, environmental-generated missiles, and site proximity missiles. The structures, shields, and barriers that are designed to withstand missile effects, the possible missile loadings, and the procedures to which each barrier is designed to resist missile impacts are discussed in this section.

3A.3.2.7.1 Missile Selection and Description

Missiles are classified as internal or external.

Externally generated missiles are those that originate from an event of natural or human-induced origin that originates outside the site and whose effects on the reactor facility are considered as potentially hazardous. Examples include environmental-generated missiles and transportation accidents.

Refer to Section 3.4.3 for internal hazards and internal missiles. A statistically significant missile is defined as a missile that could cause unacceptable plant consequences or exceedance of radiological release limits.

The examination of potential missiles and their consequences is done in the following manner to determine statistically significant missiles:

- If the probability of occurrence of the missile, P1, is determined to be less than 10^{-7} per year, the missile is dismissed from further consideration because at that likelihood it is considered not to be a statistically significant risk.
- If P1 is found to be greater than 10^{-7} per year, it is examined for its probability of impacting a design target P2.
- If the product of P1 and P2 is less than 10^{-7} per year, it is dismissed from further consideration.
- If the product of P1 and P2 is greater than 10^{-7} per year, the missile is examined for its damage probability P3. If the combined probability (i.e. P1 x P2 x P3 = P4) is less than 10^{-7} per year, the missile is dismissed.
- Finally, measures are taken to design acceptable protection against missiles with P4 greater than 10^{-7} per year to reduce P1, P2 and/or P3, so that P4 is less than 10^{-7} per year.

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

- Location of the system or component in an individual missile-proof structure.
- Provision of localised protection shields or barriers for systems or components.
- Design of the particular structure or component to withstand the impact of the most damaging missile.

The following criteria are considered as an acceptable design basis to withstand the statistically significant missiles:

- No loss of containment function as a result of missiles generated internal to containment.
- Reasonable assurance that a safe plant shutdown condition can be achieved and maintained.
- The failure of Non-Safety Class (SCN) equipment, components, or structures, whose failure could result in a missile, do not cause failure of nuclear SC equipment.
- No high energy lines are located near the off-gas charcoal bed adsorbers (located in the RWB).

In some cases, structural walls and slabs are credited for missile-consequence mitigation from missiles generated by rotating and pressurised components. These walls and slabs are designed to withstand internal missile effects to preclude perforation by internally generated missiles.

3A.3.2.7.2 Missiles Generated by Extreme Winds

The possible environmental hazards are due to missiles generated by the design basis tornado and extreme wind.

The methodology used for BWRX-300 design for missiles generated by tornadoes and other extreme winds has considered RG 1.76 (Reference 3A-70) for tornado missiles, RG 1.221, "Design Basis Hurricane and Hurricane Missiles for Nuclear Power Plants," (Reference 3A-71) for extreme wind missiles, and RG 1.143 (Reference 3A-41) missile spectrum for the RWB, along with guidance in American National Standards Institute/American Nuclear Society (ANSI/ANS)-2.3, "Estimating Tornado, Hurricane, and Extreme Straight Line Wind Characteristics at Nuclear Facility Sites," (Reference 3A-72), "Estimating Tornado, Hurricane, and Extreme Straight Line Wind Characteristics at Nuclear Facility Sites".

Design for missile impact includes the extreme winds and the corresponding wind-generated missiles listed in Table 3A-5. The design basis tornado and extreme wind missile spectrum shown in Table 3A-5 is the most limiting natural phenomena hazard in the design of missile barriers and structures. This spectrum is considered to envelope missiles generated by less intense natural phenomena.

RG 1.76 (Reference 3A-70) guidelines are followed for design basis missiles for nuclear power plants which includes at least:

- A massive high kinetic-energy missile that deforms on impact
- A rigid missile that tests penetration resistance
- A small rigid missile of a size sufficient to pass through any opening in protective barriers

Non-storm-resistant building superstructures are constructed from materials such as reinforced concrete block, or structural steel with metal siding and roof deck. Potential missiles or debris from these materials, resulting from failure of the superstructure or from items blown off, when subjected to winds of tornado or hurricane intensity, are considered to be less severe than the missiles shown in Table 3A-5.

3A.3.2.7.3 Site Proximity Missiles

As discussed in PSR Chapter 2 (Reference 3A-38), such external missiles are considered to be site-specific, with further detail provided in PSR Chapter 15.8 (Reference 3A-23).

3A.3.2.7.4 Aircraft Hazards

Considering the acceptance criteria in NUREG-0800 (Reference 3A-47) SRP 3.5.1.6, “Aircraft Hazards”, when the probability of aircraft accidents resulting in radiological consequences greater than the 10 CFR 100, “Reactor Site Criteria,” (Reference 3A-73) exposure guidelines is greater than an order of magnitude of 1E-07 per year, a detailed review of aircraft hazards is to be performed.

To mitigate their potential of equipment damage and fire impacts, the design of the BWRX-300 Seismic Category 1A structures addresses penetration resistance of buildings and considers physical separation of redundant or backup equipment, where applicable.

3A.3.2.7.5 SSCs to be Protected from Externally Generated Missiles

SSCs that are required to be protected from externally generated missiles are located in a structure that is designed as storm (including tornado) resistant or other provisions are made to protect them. BWRX-300 Seismic Category 1A and RW structures, and portions of the CB structure, defined by the different systems requirements, are designed to withstand the effects of externally generated missiles.

The design of SSCs to be protected from externally generated missiles has considered the acceptance criteria from NUREG-0800 (Reference 3A-47) SRP 3.5.2, “Structures, Systems and Components to be Protected from Externally-Generated Missiles”, ensuring that the criteria is met.

For the RWB, design for missiles is in accordance with RG 1.143 (Reference 3A-41).

The MCR and MCR to secondary control room egress route are hardened to provide a qualified route for egress, taking into consideration the requirements of RG 1.117, “Protection Against Extreme Wind Events and Missiles for Nuclear Power Plants,” (Reference 3A-74).

3A.3.2.7.6 Barrier Design Procedures

The procedures by which structures and barriers are designed to resist missiles have been developed in accordance with NUREG-0800 (Reference 3A-47) SRP 3.5.3, “Barrier Design Procedures”.

Local Damage Prediction

The prediction of local damage in the impact area depends on missile characteristics, the basic material of construction of the structure or barrier (i.e., reinforced concrete, steel, or Steel-Plate Composite or DP-SC), and the structural response. It is assumed that the missile strikes the target normal to the surface, and the axis of the missile is assumed parallel to the line of flight. These assumptions result in a conservative estimation of local damage.

Explicit dynamic inelastic analysis may be used, in lieu of the simplified formulas, to predict the local damage modes associated with the penetration of a missile into the barrier and the minimum required thickness.

Concrete Structures and Barriers

Local damage modes resulting from hard missile impact to concrete structures are presented in Figure 3A-1 adopted from R.P. Kennedy, “A Review of Procedures for the Analysis and

Design of Concrete Structures to Resist Missile Impact Effect," (Reference 3A-75), as explained below:

- Missiles with extremely low velocity can strike the target and bounce off without creating any local damage.
- As the missile velocity increases, the concrete surface is spalled. The spall area extends larger than the missile cross-sectional area as shown in Figure 3A-1(a).
- As the velocity increases, the missile can penetrate the target as shown in Figure 3A-1(a).
- Further increase in the missile velocity can result in cracking the back surface which may result in scabbing (scabbing area is much wider than that of spalling but less deep) as shown in Figure 3A-1(b).
- Further increase in the missile velocity can result in perforation of the target as shown in Figure 3A-1(c).
- Finally, with increased velocity the missile can exit from the rear side of the target.

To prevent perforation, criteria in Appendix F, Section F.7 of ACI 349 "Code Requirements for Nuclear Safety-Related Concrete Structures and Commentary," (Reference 3A-76) is satisfied as listed below:

- When perforation of concrete structural elements needs to be precluded, the concrete thickness is to be at least 20% greater than that required to prevent perforation.
- Concrete structural elements protecting required systems or equipment which could be damaged by secondary missiles (fragments of scabbed concrete) are designed to prevent scabbing, or a properly designed scab shield is based on applicable formulas or pertinent test data. In the absence of scab shields, the concrete thickness is to be at least 20% greater than that required to prevent scabbing.
- When it can be demonstrated by applicable formulas or pertinent test data that the concrete thickness is at least 20% greater than that required to prevent perforation and hence punching shear failure, design for punching shear in accordance with Section F.5 of ACI 349 (Reference 3A-76) is not required.
- For concrete slabs or walls subjected to missile impact effects where the concrete thickness is less than twice that required to prevent perforation, the minimum percentage of reinforcement is to be 0.2% each way, each face.

Empirical equations provided in R.P. Kennedy, "A Review of Procedures for the Analysis and Design of Concrete Structures to Resist Missile Impact Effects" (Reference 3A-75), are used to determine the required barrier thicknesses. Resulting thickness cannot be less than those listed in Table 3A-6, which specifies the minimum thicknesses necessary to protect against tornado missiles.

Steel Structures and Barriers

Perforation is the local damage mode considered for a hard missile impacting steel plate. The minimum plate thickness to prevent perforation is obtained by multiplying 1.25 by the more conservative thickness of the formula described in Ballistic Research Laboratory "Reactor Safeguards", and the Stanford equation in ORNL-NSIC-5, "U.S. Reactor Containment Technology," (Reference 3A-77), as recommended by Bechtel Topical Report BC-TOP-9A, "Design of Structures for Missile Impact," (Reference 3A-78).

Steel-Plate Composite Barriers

Using guidance from NUREG-0800 (Reference 3A-47) SRP 3.5.3, composite section barriers are utilised 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.

Local damage modes resulting from hard missile impact to Steel-Plate Composite structures are presented in Figure 3A-2, discussed in "Behavior, Analysis and Design of Steel-Plate Composite Walls for Impactive Loading," (Reference 3A-79), as follows:

- Stage 1: The missile can impact the front steel faceplate causing damage to the front faceplate or perforate through it.
- Stage 2: The missile creates a hole in the front faceplate (tunnelling zone) and crushes the concrete in the contact area. Radial cracks form a frustum in the concrete which begin to push the rear steel faceplate.
- Stage 3: The concrete frustum pushes against the rear faceplate causing it to deform (bulge) in the out-of-plane direction.
- Stage 4: The missile proceeds in the concrete frustum and crushes the central area. Additional smaller frustums are created within the original one and the rear faceplate bulging increases.
- Stage 5: The missile contacts the rear faceplate and if it has enough residual velocity the missile can perforate through the rear faceplate.

SC structures are designed to prevent local perforation of the rear faceplate. Scabbing is not a design limit state because it is prevented by the rear steel faceplates of the Steel-Plate Composite structure. The steel faceplate on the impact (front) side has little influence on the behaviour with the exception that it constrains concrete spalling on its side. Therefore, the front faceplate is conservatively ignored per "Design of Composite steel-plate composite Walls to Prevent Perforation from Missile Impact". The missile penetrates the concrete infill thickness and dislodges (fractures) a conical plug of concrete (frustum) that starts moving at the same residual velocity as the impacting missile. This conical concrete plug becomes the impacting projectile on the rear steel plate, and the rear steel plate needs to stop the mass of the concrete plug and original missile to prevent perforation (plate tearing/fracture) of the steel-plate composite barrier.

When the missile impact velocity is greater than the calculated perforation velocity of the concrete infill, perforation of steel-plate composite structures by missiles is prevented by specifying steel faceplate thickness that is greater than the minimum steel plate thickness. Perforation of steel-plate composite structures is not allowed; therefore, the faceplate thickness required to prevent perforation under impactive load is to be at least 25% greater than the calculated thickness.

Steel-plate composite panel thickness required for resisting missile impact is determined by calculating the missile penetration depth in the steel-plate composite panel concrete core. A modification factor is proposed, "Design of Composite Steel-Plate Composite Walls to Prevent Perforation from Missile Impact," (Reference 3A-80) to enhance the accuracy of predicting the missile penetration depth into the concrete core of steel-plate composite panels.

The missile velocity required to perforate a given wall thickness is based on the equations in NEI 07-13, "Methodology for Performing Aircraft Impact Assessments for New Plant Designs," (Reference 3A-81), modified as proposed by Reference 3A-79 and Reference 3A-80.

The minimum steel faceplate thickness required to prevent perforation of the rear steel plate by the missile and concrete plug moving together with the residual velocity is calculated per Reference 3A-80.

3A.3.2.7.7 Overall Damage Prediction

The overall response of a structure or barrier to missile impact depends upon the location of impact (e.g., near mid-span or near a support), dynamic properties of the structure/barrier and missile, and on the kinetic energy of the missile.

The overall response of a structure subjected to impactive load is determined by one of the following methods:

- Energy Balance Methods: Overall response can be assessed by using energy balance techniques to demonstrate that the calculated value of the required ductility ratio, μ_r, is less than or equal to the applicable value of the permissible ductility ratio, μ_p:
  - Williamson and Alvy "Impact Effect of Fragments Striking Structural Elements," (Reference 3A-82): An acceptable procedure for such an analysis, where the impact is assumed to be plastic, is presented. In this reference, the deformability of the missile is not considered.

BC-TOP-9A (Reference 3A-78): When a missile strikes a target, large forces develop at the missile target interface. If the interface forcing function is known (e.g., frontal impact of an automobile), the target structure can be modelled to predict the structural response. In this method, the deformability of the missile is accounted for in the forcing function. The overall structural response of a target is determined by equating the available target strain energy to the required strain energy to stop a missile using the analysis procedure provided in BC-TOP-9A (Reference 3A-78).

- Nonlinear Time-History Dynamic Analysis Method: The dynamic effects of impactive loads can be considered by performing a time-history dynamic analysis. The mass and inertial properties are included, as well as the nonlinear stiffnesses of the structural members under consideration.

3A.3.2.7.8 Dynamic Increase Factors

Dynamic increase factors are based on the material strain rate effects. Increase factors are applied to static material strengths of steel, concrete, and steel-plate composite constituents for purposes of determining section strength but not to exceed those specified in Table 3A-7.

The dynamic increase factors are limited to 1.0 for all materials where the dynamic load factor associated with the impactive loading is less than 1.2 (ACI 349 – (Reference 3A-76), Clause F.2.1 and ANSI/AISC N690-18, "Specification for Safety-Related Steel Structures for Nuclear Facilities" (Reference 3A-67), Clause Appendix N9.1.6a, "Specification for Safety-Related Steel Structures for Nuclear Facilities").

Required Ductility Ratio

Required ductility is calculated by dividing the maximum target displacement resulting from the missile impact by the component yield displacement. Maximum target displacement is estimated from the applied impact load combined with other applicable loads using one of the energy balance methods described in Williamson and Alvy (Reference 3A-82) and Energy Balance Method, BC-TOP-9A (Reference 3A-78). The component adequacy for the resulting inelastic deformation is determined by comparing the calculated required ductility ratio (μ_r) with the applicable permissible ductility ratio (μ_p) provided in Table 3A-8 for steel and reinforced concrete and Table 3A-9 for steel plate composite.

3A.3.2.8 External Fires, Explosions and Toxic Gases

3A.3.2.8.1 External Fires

The following design features minimise the probability of a fire event or assure adequate protection in the event of a fire event:

- Non-combustible and heat resistant materials such as metal cabinets, metal wireways, high melting point insulation, and flame-resistant markers for identification are used wherever practicable
- Redundant divisions of Safety Class 1 (SC1) equipment are located in separate fire areas from one another to assure that Safety Category 1 functions remain available despite a fire in any one area
- The BWRX-300 Fire Protection System (FPS) is designed and located to minimise, consistent with other safety requirements, the effects of fires and explosions
- The FPS is provided with appropriate capacity and capability to minimise impact of fires on equipment responsible for mitigating Postulated Initiating Events (PIEs), in accordance with its performance characteristics assumed in fire hazard analyses
- Location and method of fire suppression are selected such that the impacts of rupture or inadvertent operation do not prevent the performance of a Safety Category 1 function

3A.3.2.8.2 Explosions

The RB structure is designed to withstand impulsive and impactive loads as discussed in PSR Chapter 9B (Reference 3A-9), Section 9B.2.3.3.

3A.3.2.8.3 Release of Toxic Gases

Activities that could result in release of toxic gases that could impact the safe operation of the BWRX-300 are considered to be site-specific in PSR Chapter 2 (Reference 3A-38), with further detail provided in PSR Chapter 15.8 (Reference 3A-23).

Mitigation measures considered in the design of MCR/SCR are referenced in PSR Chapter 6 (Reference 3A-5).

3A.3.3 Other External Hazards

3A.3.3.1 Electromagnetic Interference

Protection against electromagnetic interference caused by lightning, high-voltage transmission lines and telecommunication towers is provided by appropriate shielding and qualification of equipment.

SC SSCs 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 PSR Chapter 8 (Reference 3A-7).

3A.3.3.2 Biological Phenomena

The Pumphouse / forebay structure will be 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 intake structure at an adequate depth in the heatsink and the installation of traveling water screens to prevent intake of biofouling material.

As shown in PSR Chapter 1 (Reference 3A-83), 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.

3A.3.3.3 Collisions of Floating Bodies and Frazil Ice with Water Intakes

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 PSR Chapter 10 (Reference 3A-10) for information related to the CWS.

3A.3.3.4 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 DBTs. 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 pond
- 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 (Reference 3A-34) further describes structures and features to detect, assess, impede, and delay threats up to and including the DBT for radiological sabotage.

3A.4 Protection Against Internal Hazards

This section discusses design basis internal hazards that could compromise the safety functions of SC1 SSCs and preventive, and mitigation measures implemented in the design to eliminate their adverse effects. Safety Class 2 (SC2) / Safety Class 3 (SC3) SSCs 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 PSR Chapters 15.5 (Reference 3A-20) and 15.6 (Reference 3A-21).

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. 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 Defence-in-Depth (D-in-D) concept discussed in Section 3.1 of PSR Chapter 3 (Reference 3A-1). They include the use of separation, barriers/shielding, and monitoring programs to preclude unacceptable radiation releases following accidents due to internal hazards.

Combination of loads from randomly occurring individual internal hazards is also considered in the design to ensure structure are adequately protected against internal hazards.

Further discussion of the internal hazards approach is provided in PSR Chapter 15.7 (Reference 3A-22).

3A.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 are discussed below.

3A.4.1.1 Internal Fires

The following design features minimize the probability and effect of fires and explosions:

- Electrical cabling is suitably rated, and cable tray loading is designed to avoid unacceptable internal heat buildup. The arrangement of equipment in reactor protection channels provide physical separation to limit the effects of fire.
- Electrical cabling supporting SC1 equipment is routed with separation from cabling supporting non-SC1 equipment.
- Separation is provided between DLs to the extent that DLs 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 fire to a single compartment or area minimizing the potential for adverse effects from fires on redundant SSC.

Protection against internal fires is provided by:

- A FPS to detect, notify, and suppress internal fires and the implementation of a comprehensive fire protection program

The FPS 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 FPS are provided in PSR Chapter 9A (Reference 3A-8).

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 and the potential consequences of design basis fire events.

A fire hazard assessment is performed as discussed in PSR Chapter 9A (Reference 3A-8) to identify the specific fire hazards and fire protection capabilities for the plant. PSR Chapter 9A (Reference 3A-8) also discusses the fire safe shutdown analysis that evaluates fire effects on the safe shutdown systems. Methodology for these evaluations is illustrated in PSR Chapter 9A (Reference 3A-8).

The D-in-D principle discussed in PSR Chapter 3 (Reference 3A-1) Section 3.1 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. Fire protection design features are discussed in PSR Chapter 9A (Reference 3A-8).

3A.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

Separation is provided between DLs to the extent that DLs 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.

3A.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.

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 PSR Chapter 9A (Reference 3A-8) for details of the BWRX-300 HVAC and FPSs, respectively.

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 PSR Chapter 6 (Reference 3A-5). As stated in PSR Chapter 6 (Reference 3A-5), 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.

For the purposes of Step 2 GDA, on-site hazard materials have been excluded from detailed assessment.

3A.4.2 Internal Flooding

Design of the plant flood protection includes all SSCs whose failure could prevent safe shutdown of the plant or result in uncontrolled release of significant radioactivity.

SSCs are also evaluated for risk effects in the internal flooding PSA using the methods described in PSR Chapter 15.6 (Reference 3A-21).

3A.4.2.1 Internal Flood Protection for On-Site Equipment Failures

The internal flood design determines the effects of discharged fluid from leaks due to pipe cracks and breaks of fluid systems. The analysis method includes an outline of the approach to show the sources of flooding, and the potential target equipment due to potential flooding or spray. The review of building layout, elevations, and rooms determines the potential list of targets that require flood protection. In the buildings where there are large areas without dividing walls, the areas are subdivided to show any source of flooding and any target equipment in each of these areas

The design criteria for protection against the effects of compartment flooding follows guidelines provided in ANSI/ANS 56.11 “Design Criteria for Protection Against the Effects of Compartment Flooding in Light Water Reactor Plants,” (Reference 3A-84), as applicable.

3A.4.2.2 Internal Flooding Evaluation Methodology

The internal flood evaluation methodology considers the flow paths associated with potential internal floods, to improve the flood protection capability and to determine the additional loads on the structure (floor and walls) associated with a particular flood event.

A structured approach is utilized to identify potential sources of flooding, affected equipment, model development, and evaluation criteria. Given the multitude of sources and equipment at risk, a series of evaluations are undertaken to identify the representative scenarios. These limiting scenarios are then used to develop mathematical models, leading to a more comprehensive and detailed analysis.

Buildings that contain SC1 SSCs are evaluated for potential flooding or areas where spray can occur. Buildings that do not contain SC1 equipment, but house systems which could cause flooding that has the potential to reach SC1 equipment through connecting low paths are also evaluated. A review of each building layout, elevation, and rooms is completed. Identification of flood area consider hydraulic communication between rooms and buildings.

Areas where potential flooding or spraying can occur in the RB are identified by reviewing the building layout, elevations, and rooms.

The piping failure criteria is provided in Section 3A.4.4.1. The internal flood evaluation includes the following sources for flooding and spray:

- Pipe ruptures:
  - Through-wall cracks in seismically supported, moderate-energy piping
  - Breaks and through-wall cracks in non-seismically supported, moderate-energy piping
  - High-energy piping (Breaks and Cracks)
- Tank ruptures
- Overfill of tanks
- Pump seals
- Fire suppression piping
- Fire sprinklers
- Piping ruptures to/from tanks located in the outside area

The potential sources of water that can cause internal flooding are evaluated to determine possible local effects, such as water height and flow. Within this affected region, SC1 equipment are identified.

The necessary SC1 equipment are evaluated for operability against the most limiting environmental conditions for which they are to be located in.

If the equipment is deemed to be inoperable, protective features are used to mitigate or eliminate the consequences of internal flooding. The mitigating strategy applied to the design is intended to ensure the equipment is compatible with the environmental conditions. If required, design features such as drainage or sump pumps are used to mitigate the effects of flooding. Possible mitigating strategies also include crediting operator actions, physical relocation, mitigating hardware, or environmental qualification (submergibility).

3A.4.3 Internal Missiles

Internally generated missiles are those resulting from in-plant equipment component failures or other events within the nuclear island. Potential missile sources from rotating equipment, high-pressure system ruptures, and missiles caused by, or because of, gravitational effects (e.g. heavy load drop) are considered, when applicable. This understanding is derived from NUREG-0800 (Reference 3A-47), SRP 3.5.1.1, "Internally-Generated Missiles (Outside Containment)", and NUREG-0800 (Reference 3A-47) SRP 3.5.1.2, "Internally-Generated Missiles (Inside Containment)".

Missile selection is discussed in in Section 3A.3.2.7.

Preventative and mitigative measures considered in the BWRX-300 design against external missiles have been listed in Section 3A.3.2.7, and can also be applicable for internal missiles, with the following additions:

- Physical separation of redundant systems or components in an individual missile-proof structure
- Provision of design features on the potential missile source to prevent missile generation
- Orientation of the potential missile source to prevent unacceptable consequences caused by missile generation

3A.4.3.1 Internally Generated Missiles (Outside Containment)

Potential missile sources from rotating equipment or high-pressure system ruptures, and missiles caused by, or because of, gravitational effects (e.g., heavy load drop) are considered in the design, where applicable.

Equipment within the general categories of pumps, fans, blowers, diesel generators, compressors, and components in systems normally functioning during power reactor operation, are examined for possible sources of credible and significant missiles.

Potential missiles that result from the failure of pressurized components are categorised as contained fluid energy missiles or stored energy (elastic) missiles.

Examples of potential contained fluid energy missiles are valve bonnets, valve stems, and retaining bolts. Valve bonnets are considered jet-propelled missiles. Valve stems are analysed as piston-type missiles, while retaining bolts are examples of stored strain energy missiles.

BWRX-300 does not consider pressurized bottles in the design.

3A.4.3.2 Internally Generated Missiles (Inside Containment)

SSCs (inside containment) are protected from internally generated missiles to ensure support to the reactor facility. Potential missile sources from both rotating equipment and pressurized components are considered, when applicable. Gravitational missiles inside the containment are also considered.

3A.4.3.3 Turbine Missiles

The risk from turbine missiles will be shown to be acceptably small, either because design features are provided to prevent damage or because the probability of a strike by a turbine missile is sufficiently low. PSR Chapter 15.7 (Reference 3A-22) provides further discussion on the treatment of turbine missiles, recognising that internal missiles and turbine missiles were screened out from deterministic assessment based on probabilistic arguments.

Favorable turbine generator placement and orientation, combined with quality assurance in design and fabrication, maintenance, and inspection programs, and overspeed protection systems, as described in PSR Chapter 10 (Reference 3A-10), provide an acceptably small risk from turbine missiles. If the risk is acceptably low, barriers are not required to protect SC1 items. Turbine missile characteristics, trajectory paths, and probabilities are specific to each turbine vendor and turbine type; therefore, detailed turbine missile data is provided after selection of a turbine vendor. PSR Chapter 15.6 (Reference 3A-21) presents the methodology for performing a preliminary qualitative assessment for determining the probability of turbine generated missiles and their effect on target SSCs.

3A.4.4 Dynamic Effects Associated with the Postulated Rupture of Piping

The plant is designed for protection against piping failures inside and outside containment to assure that such failures do not cause the loss of needed functions of the systems and to ensure that the plant can be safely shut down in the event of such failures. The design includes consideration of high energy and moderate energy fluid system piping located inside and outside of containment. This section discusses the design bases and measures used to protect essential SSCs. Essential SSCs are those that are required to shut down the reactor and mitigate the consequences of the postulated piping failure. In addition, any systems, or portions of systems, that are designed to mitigate the consequences of a postulated pipe failure are provided with design features that can assure the performance of their safety function.

3A.4.4.1 Plant Design for Protection Against Postulated Piping Failures in Fluid Systems Inside and Outside Containment

3A.4.4.1.1 Design Basis

USNRC BTP 3-3 “Protection Against Postulated Piping Failures in Fluid Systems Outside Containment,” (Reference 3A-85), and BTRP 3-4, “Postulated Rupture Locations in Fluid System Piping Indie and Outside Containment,” (Reference 3A-86), have been used as the design bases and criteria for the analysis required to demonstrate that essential SSCs are protected from piping failure. BTP 3-4 (Reference 3A-86) describes criteria for selecting the locations and orientations of postulated breaks and cracks in fluid systems piping.

Break Exclusion Zone

The break exclusion zone typically runs from the inboard Containment Isolation Valve (CIV) through the containment penetration to the outboard CIV as described in BTP 3-4 B.1(ii) (Reference 3A-86).

In the BWRX-300 design, the Reactor Isolation Valve (RIV) is acting as the inboard CIV for all high energy systems. The boundaries of the break exclusion zone run from the RIV (inboard CIVs) to the outboard CIVs and past the outboard CIVs to the Seismic Interface Restraints (SIR).

NEDC-33911P-A, “BWRX-300 Containment Performance,” (Reference 3A-87), Section 5.1.7 discusses the pipe break exclusion zone boundaries where breaks and cracks are postulated in those portions of piping from the RIVs that function as the inboard CIVs to the containment wall. NEDC-33910P-A, “BWRX-300 Reactor Pressure Vessel Isolation and Overpressure Protection,” (Reference 3A-88) discusses that each RIV assembly is also connected to the outboard piping using bolted flange connections.

This description of the Break Exclusion Zone reflects the standard design at the UK design reference point in 2024 (see PSR Chapter 3 (Reference 3A-1)). UK-specific application and consideration of the break exclusion zone is discussed further in PSR Chapter 15.7 (Reference 3A-22), and is subject to FAP PSR15.7-63.

Augmented Inservice Inspections

As specified in BTP 3-4 B.1(ii)(7) (Reference 3A-86) and ASME Code, Section XI, “Rules for Inservice Inspection of Nuclear Reactor Facility Components,” (Reference 3A-89), IWA-2400, a 100% volumetric in-service examination of all pipe welds should be conducted during each inspection interval. The welds and base metal in the break exclusion zone are pre-examined and in-service inspected in accordance with ASME BPVC-XI, BPVC Section XI (Reference 3A-89). The ISI is performed on 100% of the welds during each inspection interval.

The break exclusion zone piping is assessed for the potential of damage mechanisms other than weld cracking. The break exclusion zone specific damage mechanisms are identified, based on industry data, including using the latest revisions of:

- USNRC NUREG-1801 “Generic Aging Lessons Learned,” (Reference 3A-90)
- USNRC NUREG-2191 “Generic Aging Lessons Learned for Subsequent License Renewal,” (Reference 3A-91)

The break exclusion zone piping is systematically assessed for all the potentially applicable Generic Aging Lessons Learned damage mechanisms (NUREG-1801 and NUREG-2191), and Operational Experience (OPEX) such as published by EPRI, and the corresponding periodic inspections are developed. These augmented inspections are an essential part of assuring that piping in the break exclusion zone is not subjected to leakage or rupture. The ISI are implemented using Non-Destructive Examination techniques that have been demonstrated and qualified to detect threshold values of metal loss, cracking, or embrittlement as part of the break exclusion zone ISI procedure. The break exclusion zone piping is designed and laid out to provide access for ISIs of every pipe segment, bend, and weld. The break exclusion zone piping is subjected to augmented pre-operational testing for evidence of vibration, in accordance with ASME “Operation & Maintenance (OM) of Nuclear Power Plants,” (Reference 3A-92).

Protection of Essential Structures, Systems and Components

The design of the plant provides adequate general protection against postulated high energy and moderate energy fluid system pipe ruptures, to the extent necessary to ensure the integrity and operability of essential SSCs required to safely shut down the plant and mitigate the consequences of the postulated piping failure in accordance with regulatory requirements. Both direct dynamic effects on essential SSCs inside the zone of influence and indirect dynamic effects are considered. Examples of an indirect effect would be the whipping pipe impacting a structural frame to which essential SSCs are attached, dislodged grating resulting in a secondary missile, or missiles from back face scabbing or front face spalling on reinforced concrete caused by pipe impact.

If the consequences of dynamic effects on essential SSCs is unacceptable, the following mitigation options apply:

- The essential SSCs are relocated
- The whipping pipe is restrained by a whip restraint, resulting in a smaller restrained zone of influence that does not encompass the essential target
- A protective barrier or jet shield is used to protect the essential target from impact (barrier) or jet impingement (barrier or jet shield)
- The essential SSCs are qualified to operate following the pipe rupture interaction

3A.4.4.1.2 Definition of High Energy and Moderate Energy Fluid Systems

High energy fluid systems are defined as those systems or portions of systems that, during normal plant conditions are either in operation or are maintained pressurised under conditions where either or both of the following are met:

- Maximum operating temperature exceeds 95°C (200°F)
- Maximum operating pressure exceeds 1900 kPa (275 psig)

Systems or portions of systems pressurized above atmospheric pressure during normal plant conditions that are not classified as high energy are classified as Moderate Energy Fluid Systems.

Through-wall leakage cracks in lieu of breaks are postulated in the piping of those fluid systems that qualify as high energy fluid systems for only a short operational period but qualify as moderate energy fluid systems for the major operational period. An operational period is considered “short” if the fraction of time that the system operates within the pressure-temperature conditions specified for high energy fluid systems is less than 2% of the time that the system operates as a moderate energy fluid system. PSR Chapter 15.7 (Reference 3A-22) provides further detail and discusses the application of this definition.

3A.4.4.1.3 Design Safety Evaluation

Essential SSCs are protected against postulated piping failures. The effects of postulated breaks and cracks include pipe whipping, jet impingement, fluid decompression waves within the ruptured pipes and environmental effects such as temperature, pressure, humidity, chemical exposure, radiation, spray wetting, flooding, and sub compartment pressurization.

An evaluation is performed to verify that the consequences of failures of high and moderate energy lines do not affect the ability to safely shut down the plant.

3A.4.4.2 Determination of Rupture Locations and Dynamic Effects Associated with the Postulated Rupture of Piping

NUREG-0800 (Reference 3A-47), SRP 3.6.2 has been used to consider the break and crack location criteria and methods of analysis for dynamic effects. This includes 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.

3A.4.4.2.1 Analytical Methods to Define Forcing Functions and Response Models

Analytical Methods to Define Blowdown Forcing Functions

Thrust force and its magnitude is determined at the postulated circumferential and longitudinal break are developed by analytical solutions (closed-form equations) or by numerical simulation (computational fluid dynamics techniques). ANSI/ANS 58.2, “Design Basis for Protection of Light Water Nuclear Power Plants Against the Effects of Postulated Pipe Rupture,” (Reference 3A-95), modified by NUREG/CR-7275, “Jet Impingement in High Energy Piping Systems,” (Reference 3A-96), has been considered in developing the analytical methods used to establish pipe rupture blowdown and jet thrust forcing functions.

High Energy Pipe Whip Analysis

The analysis of pipe whip follows analytical methods using dynamics equations or a numerical method using computational fluid dynamics and/or finite element analysis. The following are some of the considerations that apply to pipe whip analysis:

- Two ends of the broken pipe move clear of each other unless physically limited by piping restraints, structural members, or piping stiffness.
- Determine the direction of the thrust force, axial to the broken pipe segment (perpendicular to the break area).
- The unrestrained zone of influence is determined in the direction of the jet reaction initially, with the total path controlled by the piping geometry.
- If unrestrained, a whipping pipe having a constant energy source sufficient to form a plastic hinge is considered to form a plastic hinge and rotate about the plastic hinge or the nearest rigid pipe whip restraint, anchor, or wall penetration capable of resisting the pipe whip loads; or rotate about the calculated dynamic plastic hinge location.
- Where the hinge moment has a torsional and a bending component, the hinge does not result in a planar unrestrained zone of influence. In this case, a half-sphere

unrestrained zone of influence is assumed with the pipe sweeping a 180° arc from its initial position to its final resting position, unless a more detailed analysis is performed to justify a smaller unrestrained zone of influence.

- If the plane of motion of a whipping pipe is normal to a flat surface at impact, it is assumed that the pipe comes to rest against that surface.
- The internal fluid energy level associated with the pipe break reaction considers any line restrictions (e.g., flow limiter) between the pressure source and break location, and the effects of either single-ended or double-ended flow conditions, as applicable. The energy level in a whipping pipe is considered insufficient to rupture an impacted pipe of equal or greater nominal pipe size and equal or heavier wall thickness.
- Credit is taken for the limited amount of energy in closed ended piping runs near dead ends or normally closed valves.
- For those portions of high energy piping systems that are normally pressurized only during plant conditions other than 100 percent power, the thermodynamic state that produces the most severe fluid reaction forces is used.

Dynamic Analysis Methods to Verify Integrity and Operability

Jet Impingement Analyses

Guidance from ANSI/ANS 58.2 (Reference 3A-95) has been used for modelling of the jet geometry and calculation of the jet impingement force acting on a target, with modifications applied as identified in NUREG/CR-7275 (Reference 3A-96).

Design Codes and Load Combinations for Pipe Whip Restraint

Pipe whip restraints are BWRX-300 Seismic Category 2. They are seismically designed to prevent adverse interaction with BWRX-300 Seismic Category 1A and 1B SSCs. Pipe whip restraints are designed to the following codes and standards:

- Pipe whip restraints that only act to restrain the effects of pipe breaks, and are not active for other loads, are designed to ANSI/American Institute of Steel Construction (AISC) N690-18 "Specification for Safety-Related Steel Structures for Nuclear Facilities" (Reference 3A-67). They are not ASME III NF restraints.
- Pipe whip restraints that also act as restraints for other loads (e.g., deadweight, seismic) are ASME BPVC-III-1-NF "Rules for Construction of Nuclear Facility Components Division 1 – Subsection NF Supports," (Reference 3A-97) restraints. However, the qualification for the pipe break impact load follows ANSI/AISC N690-18 (Reference 3A-67).
- The pipe whip restraints are designed to allow access for ISIs. The measures taken for the protection of essential SSCs do not preclude the conduct of in-service examinations of ASME Class 2 and 3 pressure-retaining components as required by the rules of ASME Boiler and Pressure Vessel Code - Section XI.
- The pipe-restraint gap is sufficiently small to limit the kinetic energy of the whipping pipe and therefore the impact load.
- The pipe-restraint gap is sufficiently large to prevent interference from thermal expansion of the line.
- Pipe whip restraints are generally located to prevent the formation of a plastic hinge.

The pipe whip restraints, jet shields, and barriers are designed for the following load and load combinations (Service Level D):

$$DW + TH + \sqrt{DBPB^2 + (|SSE| + |SSES| + |SWE|)^2}$$

Where:

DBPB = Design Basis Pipe Break (including jet impingement)

DW = Deadweight

SSE = Safe Shutdown Earthquake

SSES = SSE seismic anchor motion of the restraint structure

SWE = Seismic Self-Weight excitation of the restraint structure

TH = thermal loads due to the expansion of the whip restraint relative to the post-break hot building structure.

Analytical Methods to Define Blast Waves Interaction to Essential SSCs

Blast Wave Effects

- High energy blast wave analysis is performed assuming the break opening time as instantaneous, maximizing blast formation.
- The formation and effects of a blast wave caused by a High Energy Line Break is evaluated using three-dimensional computational fluid dynamics modelling technique that reflects the thermohydraulic parameters at the instant of the postulated pipe rupture, within 1 millisecond of the onset of the break.
- These blast wave forces on target surfaces are impulsive loads that last a few milliseconds or less.
- The blast analysis includes amplification caused by blast wave reflections against walls, and the effects of angled incidence.
- Because of pressure-relieving clearing at the edges of a target surface, and the short duration of the pressure impulse, small structures are not exposed to significant loading. The analysis of blast wave effect is therefore limited to walls, floors, and large vessels.
- Blast force and impulse may be bounded by the jet thrust forces that subsequently develop.

Pipe Break Analysis Results and Protection Methods

A summary of the dynamic analyses applicable to high energy piping systems follows the guidelines provided in RG 1.70, “Standard Format and Content of Safety Analysis Reports for Nuclear Power Plants,” (Reference 3A-98). This information is provided and completed in conjunction with the operating license application process.

3A.4.5 Other Internal Hazards

3A.4.5.1 Hard Object Impact

Complying with IAEA SSG-64, “Safety Standards – Protection against Internal Hazards in the Design of Nuclear Power Plants,” (Reference 3A-99), the BWRX-300 design considers hard object impact loads resulting from the drop of heavy loads lifted and handled in areas where SSCs 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.

In accordance with USNRC RG 1.244, "Control of Heavy Loads at Nuclear Facilities," (Reference 3A-100), the BWRX-300 heavy load is defined per the provisions of NUREG-0612 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 USNRC RG 1.244 (Reference 3A-100) and Section 5.1 of NUREG-0612, "Control of Heavy Loads at Nuclear Power Plants: Resolution of Generic Technical Activity A-36," (Reference 3A-101). 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 NUREG-0612 (Reference 3A-101).

PSR Chapter 9A (Reference 3A-8) 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.

PSR Chapter 9A (Reference 3A-8) 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, "Cranes, Rules for Construction of Overhead and Gantry Cranes, (Top Running Bridge, Multiple Girder)," (Reference 3A-102). The refuelling 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.

PSR Chapter 15.7 (Reference 3A-22) provides further detail on dropped loads, recognising that the hazard is screened out on probabilistic grounds in the Deterministic Safety Assessment.

3A.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 BWRX-300 Seismic Category 1A or 1B, or 2 SSCs in the event of DBE.

Architectural components and shielding blocks whose failure or dislocation could affect the safe operation of any BWRX-300 Seismic Category 1A or 1B, or 2 SSCs 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.

PSR Chapter 15.7 (Reference 3A-22) provides further detail on the failure of non-structural elements, recognising that the hazard is again screened out on probabilistic grounds.

3A.4.5.3 Electromagnetic Interference

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

SC SSCs 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 Section 3.9 of PSR Chapter 3 (Reference 3A-1).

Plant grounding, lightning protection and electromagnetic compatibility systems and their design requirements are discussed in PSR Chapter 8 (Reference 3A-7).

PSR Chapter 15.7 (Reference 3A-22) recognises that the electromagnetic interference hazard is not considered in for Step 2 GDA and will be assessed in future work.

3A.5 Design of Civil Structures

This Section presents the general design principles, general design basis requirements and general criteria used in the design of the BWRX-300 civil structures, including their foundations.

Below are the key sections that impact the BWRX-300 civil engineering and structural design that should be reviewed along with this section:

- PSR Chapter 1 (Reference 3A-83) – Introduction and General Considerations, which describes the generic BWRX-300 design and how it could be constructed, operated, and decommissioned in the United Kingdom (UK) on a site bounded by the generic site envelope in a way that is safe, secure and that protects people and the environment.
- PSR Chapter 2 (Reference 3A-38) - Site Characteristics, which details the site characteristics and their future evaluation in support for the design, safety assessment and periodic safety review of the BWRX-300. These characteristics are inputs for the generic design of the BWRX-300 Structures, Systems, and Components (SSCs) for any specific candidate site in the United Kingdom (UK).
- PSR Chapter 3 (Reference 3A-1) - Safety Objectives and Design Rules for Structures, Systems and Components, Section 3.1, which provides the general design aspects and D-in-D safety framework utilised in the BWRX-300 design.
- PSR Chapter 3 (Reference 3A-1) - Safety Objectives and Design Rules for Structures, Systems and Components, Section 3.2, which provides the general classification of BWRX-300 SSCs, and the approach used to establish these classifications.
- PSR Chapter 9B (Reference 3A-9) - Civil Structures, which provides specific information on compliance with the design rules for civil engineering works and structures.

In addition, Sections 3A.3 and 3A.4 provides methodology and general design requirements for protection against the effects of external and internal hazards.

From the site layout presented in 007N7334 (Reference 3A-55), the primary buildings in the BWRX-300 Power Block consist of the Reactor Building (RB) which houses the containment, RWB, Control Building (CB), TB, Service Building (SB) and Reactor Auxiliary Structures (RAS). In the following sections, reference to the integrated RB structure is inclusive of the RB, Steel-Plate Composite Containment Vessel (SCCV), and containment internal structures, whereas RB is used to refer to the part of the integrated structure located outside of the SCCV.

The seismic categorisation of these structures is provided in Table 3A-1. Per PSR Chapter 3 Section 3.2.3 (Reference 3A-1) and Table 3A-1, the Seismic Category 1A integrated RB housing Safety Class 1 SSCs has the utmost importance to safety and is credited for the safety analysis of the BWRX-300. Other civil structures for which nuclear safety relevant design basis requirements are required are the Pumphouse/Forebay structures and tunnels that support the condenser cooling and plant cooling water systems. These structures are however of lower maturity as they are site specific and therefore design basis details are not included in the current revision of this preliminary safety report. All remaining BWRX-300 structures are not credited in the safety analysis.

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 1A integrated RB. Design principles for the RWB, CB, TB, SB, and RAS, are provided in PSR Chapter 9B (Reference 3A-9) Section 9B.3.

The remaining plant structures of the BWRX-300 are not covered since they are not credited in the safety analysis.

3A.5.1 General Design Principles – Structural and Civil Engineering

3A.5.1.1 General Structural Philosophy

Based on lessons learned from the AP1000 units, DP-SC modules are being utilised for the RB. DP-SC units provide modularity and off critical path construction capability. DP-SC modules also eliminate the foundation to wall interface challenges experienced with the AP1000 units and provide a construction solution that ensures both structural integrity and light-tightness. Structural integrity is ensured via the diaphragm plates by preventing delamination of the concrete core. Additionally, the diaphragm plates provide composite action between the steel faceplates and the concrete infill, and act as out-of-plane shear reinforcement for the composite section.

3A.5.1.2 Safety Functional Requirements

The design maturity of the RB at GDA Step 2 remains in development, with aspects of the structure based on OPEX and high-level inputs from early standard design safety studies. Furthermore, both Deterministic Safety Analyses (DSA) and Probabilistic Safety Analyses (PSA) are in development, therefore the safety requirements are not final and shall be iterated on as the safety analysis matures. To support GDA step 2 NEDC-34354P, "BWRX-300 Generic Design Assessment (GDA) Reactor Building Civil Structures Safety Requirement Schedule" (Reference 3A-148) presents the current status of Safety Requirements (SRs) which address the Fundamental Safety Properties (FSPs), and ultimately the Fundamental Safety Functions (FSFs) applicable to Seismic Category 1A structures, and is intended to be an exemplar of UK regulatory norms for future delivery stages as identified in FAP item 9B-313 (See Appendix B of PSR Chapter 9B (Reference 3A-9)).

3A.5.1.3 General Design Approach

The BWRX-300 Seismic Category 1A 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 Design Basis Accident (DBA) in compliance with the requirements set in 10 CFR 50 (Reference 3A-39) GDCs 2, 4 and 16.

The RWB is classified as High Hazard (RW-IIa) per USNRC RG 1.143 (Reference 3A-41) and is similarly designed to meet the serviceability, strength, and stability requirements for load combinations under the categories of normal operation, AOO, and DBA that are relevant to that building. In addition to design requirements, requirements for fabrication, construction, examination, and testing of all structures and their foundations ensure a level of safety and performance commensurate with international Relevant Good Practice (RGP).

Each of the other structures are designed in accordance with rules detailed in local building codes and standards.

3A.5.1.3.1 Standard Design Approach

As described in NEDC-34154P, "BWRX-300 UK Generic Design Assessment (GDA) Design Reference Report," (Reference 3A-152) the Design Reference for the BWRX-300 UK GDA is based on the physical design progression of the standard plant design as of the end of March 2024. The meaning of the phrase "Standard Plant" is dependent on the BL, which is being referred to, and refers to both generic and site-specific design. This section clarifies the meaning of this phrase as it pertains to each BL:

- BL0 design phase: The “Standard Plant” requirements established and high-level conceptual design developed.
- BL1 design phase: The generic "Standard Plant" design for civil structures is developed based on an envelope of generic subgrade profiles representative of the candidate sites across North America as described in 3A.3.1.1.1.
- BL2 design phase: The detailed “Standard Plant” design for civil structures is developed based on site-specific requirements and characteristics of a particular site.
- BL3 design phase: The final "Standard Plant" design for civil structures which is specific to a particular site and may incorporate customer-specific changes to design documents.

The purpose of the Standard Plant design is to ensure repeatability of the design in multiple geographies. The Standard Design bounds a wide range of realistic seismological and geotechnical conditions that exist at most US candidate sites. At the Design Reference point for the BWRX-300 UK GDA , the integrated RB has reached BL1 progressing to BL2 design development, whilst the remaining Power Block structures (i.e. RWB, CB, TB) are of a lower design maturity. For each local site, the Standard Design will be tested against local site hazard and ground conditions to substantiate the design in that specific location.

3A.5.1.4 General Codes and Standards Approach

The current BWRX-300 civil engineering generic design, called the Standard Design, is based primarily on a US-centric design basis approach, US codes and standards and Regulatory Guidance.

The integrated RB structure and its common foundation are primarily constructed using an advanced Steel-Plate Composite system. This system has a configuration similar to the typical Steel-Plate Composite system except that the tie-rods are replaced by diaphragm plates. This type of Steel-Plate Composite system is referred to as a Diaphragm Plate Steel-Plate Composite (DP-SC) modular system. Further details on the DP-SC modules are provided in PSR Chapter 9B, Section 9B.2 (Reference 3A-9).

The design of the Steel-plate Composite Containment Vessel (SCCV) serving as the containment pressure boundary is performed in accordance with the provisions detailed in Section 6 of DPSC and Reactor Building Design Report (Reference 3A-57). US codes and standards underpin the DPSC and Reactor Building Design Report (Reference 3A-57) design basis since there are no equivalent standards or regulatory guidance in the UK.

Internal to the containment are shielding structures and steelwork structures that structurally support the containment pressure vessel, cooling systems and pipework and other safety systems. The Class MC containment metal components which compose the containment closure head, the two containment airlocks and the containment penetrations, are designed in accordance with the provisions of ASME BPVC, Section III, Subsection NE, “Rules for Construction of Nuclear Facility Components,” (Reference 3A-104).

The RWB is designed in accordance with USNRC RG 1.143 (Reference 3A-41).

Eurocodes are used for the design of the other structures with the exception of the lateral force resisting system under seismic and wind loads. These require the use of nuclear specific codes such as ACI 349 (Reference 3A-76) to limit the risk of interaction with BWRX-300 Seismic Category 1A or 1B SSCs (i.e. the RB).

3A.5.1.5 Loads and Load Combinations

The loads and load combinations that are considered in the BWRX-300 civil structures' design are in line with the codes and standards that are relevant to the structure in question and ensure that structures are designed to remain robust under applicable hazard withstand demands as required by nuclear safety requirements.

3A.5.1.6 Design Life

Overall plant design life shall be 60 years of power operation plus an allowance for decommissioning. Fatigue consideration, when applicable, shall be assessed for the specified design life. Serviceability of the structure, long term deflections, and settlement considerations shall also be assessed for the specified design life according to the applicable codes and regulatory guides.

3A.5.1.7 Beyond Design Basis and Robustness Measures

The robustness of the design to prevent potential release of radioactivity to the public and environment under Design Extension Conditions (DECs) is considered for the SCCV and RB, as discussed in PSR Chapter 9B (Reference 3A-9) Section 9B.2.6. A lack of cliff edge is demonstrated through an understanding of load paths and failure modes, a consideration of appropriate design and detailing of all structural forms and types present within the structures, and through margins assessments deploying a high confidence of low probability of failure approach.

The RB wall is designed to withstand a large aircraft impact with no penetration and SSC located in multiple Power Block building (including the RB) are credited to maintain fuel cooling from a small aircraft impact. Therefore, the RB in isolation can be credited to maintain fuel cooling and prevent core damage under any beyond design basis aircraft impact.

For all the other BWRX-300 structures, beyond design basis and DECs are not explicitly considered.

3A.5.2 BWRX-300 Civil Structures

This section presents high-level discussion with reference to PSR Chapter 9B (Reference 3A-9) on the design rules for the foundation supporting the integrated RB structure.

3A.5.2.1 Foundations

The deeply embedded cylindrical integrated RB structure is founded on a common circular mat foundation that supports the SCCV, containment internal structures and RB. This common mat foundation is made of DP-SC modules that are of the equivalent BWRX-300 Seismic Category 1A to the supported superstructures.

A complete structural description of the integrated RB common mat foundation is described in PSR Chapter 9B (Reference 3A-9) Section 9B.1.2. More detail on the other structure foundations is presented in PSR Chapter 9B (Reference 3A-9) Section 9B.1.2.

3A.5.2.2 Integrated Reactor Building

3A.5.2.2.1 Containment

The BWRX-300 containment comprises a Steel-plate Composite Containment Vessel (SCCV), a steel containment closure head and other Class MC components. The SCCV is constructed of DP-SC modules.

The SCCV, closure head and other Class MC components are designed with cognizance of their structural role and safety function requirements as described in PSR Chapter 9B (Reference 3A-9), Section 9B.2.1.1.

A complete structural description of the containment structures; SCCV, closure head and other MC components is described in PSR Chapter 9B (Reference 3A-9), Section 9B.2.1.2.

3A.5.2.2.2 Containment Internal Structures

The BWRX-300 containment internal structures comprise the DP-SC RPV pedestal, the steel-plate composite bioshield surrounding the RPV pedestal and the structural steel Containment Equipment and Piping Support Structure and two support platforms as shown in PSR Chapter 9B (Reference 3A-9), Figure 9B-1.

The containment internal structures are designed with cognizance of their structural role and safety function requirements as described in PSR Chapter 9B (Reference 3A-9), Section 9B.2.2.1.

A structural description of the containment internal structures is described in PSR Chapter 9B (Reference 3A-9), Section 9B.2.2.2.

3A.5.2.2.3 Reactor Building Outside Containment

The RB is a deeply embedded cylindrical shaped building made of DP-SC floors and walls, with steel beams supporting the roof.

The below grade portion of the RB structure encloses the containment and protects the RPV, reactor support and safety systems, and the majority of vital and non-vital power supplies and equipment. The above grade portion of the RB structure houses the refuelling floor, refuelling and fuel handling systems, fuel pool, and polar crane.

The RB is designed with cognizance of its structural role and safety function requirements as described in PSR Chapter 9B (Reference 3A-9), Section 9B.2.3.1.

A complete structural description of the RB is described in PSR Chapter 9B (Reference 3A-9), Section 9B.2.3.2.

3A.5.2.2.4 Reactor Building Pools and Liners

The BWRX-300 RB pools and liners are located in the upper part of the RB. There are 9 pools in total, as discussed in PSR Chapter 9B (Reference 3A-9), Section 9B.2.4.

The building pools and liners are designed with cognizance of their structural role and safety functional requirements as described in PSR Chapter 9B (Reference 3A-9), Sections 9B.2.4.1.

A structural description of the pools and liners is provided in PSR Chapter 9B (Reference 3A-9), Section 9B.2.4.2.

3A.5.2.3 Radwaste Building

The RWB is a largely concrete building with interior steel columns and a steel-concrete composite roof housing a portion of the Off-gas System (OGS), including the charcoal adsorbers, refueling water storage tanks, and rooms and equipment for handling, processing, and packaging liquid and solid radioactive wastes.

The RWB is designed with cognizance of its structural role and safety function requirements as described in PSR Chapter 9B (Reference 3A-9), Section 9B.3.1.1.

A complete structural description of the RWB is described in PSR Chapter 9B (Reference 3A-9), Section 9B.3.1.2.

3A.5.2.4 Control Building

The CB has reinforced concrete shear wall perimeter, interior steel columns, beams/girders, and a steel-concrete roof deck. The CB houses the MCR and electrical, control and instrumentation equipment. The CB also houses a qualified evacuation route between the MCR and the SCR which is housed within the RB.

The CB is designed with cognizance of its structural role and safety function requirements as described in PSR Chapter 9B (Reference 3A-9), Section 9B.3.2.1.

A complete structural description of the CB is described in PSR Chapter 9B (Reference 3A-9), Section 9B.3.2.2.

3A.5.2.5 Turbine Building

The TB encloses the turbine generator, main condenser and auxiliaries portions of the condensate and feedwater systems, exciter and isophase bus ducts, off-gas system cooler, the condensate filters and demineralisers, bridge crane and other systems.

The TB is comprised of 3 separate structural systems founded on the same raft:

- The TB shell structure, which consists of a steel frame system with steel columns, beams/girders, roof bar joists, and floor/roof decks as gravity load carrying systems.
- The TB Shield Wall Area (SWA) consists of a perimeter reinforced concrete shear wall and floors.
- The reinforced concrete pedestal supporting the turbine, generator, and exciter.

The TB is designed with cognizance of its structural role and safety function requirements as described in PSR Chapter 9B (Reference 3A-9), Section 9B.3.3.1.

A complete structural description of the TB is described in PSR Chapter 9B (Reference 3A-9), Section 9B.3.3.2.

3A.5.2.6 Service Building

This building is of low maturity with some outline details presented in PSR Chapter 9B (Reference 3A-9), Section 9B.3.4.

3A.5.2.7 Reactor Auxiliary Structures

This building is of low maturity with some outline details presented in PSR Chapter 9B (Reference 3A-9), Section 9B.3.4.

3A.5.2.8 Fire Water Storage Tank and Pump Enclosure

This building is of low maturity with some outline details presented in PSR Chapter 9B (Reference 3A-9), Section 9B.3.4.

3A.6 Mechanical Systems and Components – Supplementary Information

Section 3A.6 provides the general design aspects used for SC and Non-Safety Class (SCN) mechanical systems and components. It includes special considerations for mechanical components, dynamic testing, and analysis of SSCs, required codes for ASME BPVC Section III, Division 1, Class 1, 2, and 3 components, and Subsection NF for component supports, and Subsection NG for core support structures. In addition, general design aspects for CRD system, and reactor vessel internals are presented. Further, this section discusses the functional design, qualification, and In-Service Testing (IST) program requirements for pumps, valves, and dynamic restraints.

The general design principles, criteria, and classification used for design of mechanical systems and components have been discussed earlier in PSR Chapter 3 (Reference 3A-1). 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 that is commensurate with the safety classification and seismic 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.

Section 3A.6 develops the seismic input criteria and building spectra used as input for seismic qualification of BWRX-300 Seismic Category 1B active mechanical components and system functionality. Additionally, BWRX-300 Seismic Category 1A passive mechanical component supports, and equipment supports use the seismic spectra for qualification.

Equipment qualification requirements are provided in PSR Chapter 3 (Reference 3A-1), Section 3.9 for seismic and dynamic qualification of mechanical and electrical equipment, and provides the equipment qualification requirements including environmental, functional qualification, and electromagnetic compatibility, which are used as input to SC mechanical systems and components.

Codes and Standards Used in the Design of Mechanical Systems and Components

ASME BPVC, Section III, Division 1, “Rules for Construction of Nuclear Facility Components – Appendices,” (Reference 3A-105), ASME B31.1, “Power Piping,” (Reference 3A-106), and ASME B31.3, “Process Piping,” (Reference 3A-107), are applied for the design of mechanical systems, components and piping including piping components. Table 3A-10 provides the pressure boundary codes and standards utilised in the BWRX-300 mechanical system and component design.

Mechanical Equipment Separation for Safety Class 1 Equipment

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 SC and SCN piping, valves, and associated mechanical controls, and instrumentation. Additionally, adjacent systems are considered in mechanical equipment separation (as related to human factors, mechanical maintenance, and seismic interaction).

Principles of physical separation include:

- Separation by geometry (layout, distance, orientation, elevation, and, including separate structures)
- Separation by barriers (e.g., walls, shields), both vertical and horizontal
- Separation by a combination of geometry and barriers

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.

The redundancy methods are used to protect from single active failures or events; examples include:

- Use of SC structures
- Spatial separation
- Three-hour rated fire barriers
- Isolation devices
- Redundancy of instrumentation

The application of the single failure criterion is described in NEDC-33934P "BWRX-300 Safety Strategy Licensing Topical Report," (Reference 3A-108).

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.

The following SC and SCN SSCs mechanical equipment items are considered for separation:

- Piping systems
- Valves
- Rotating equipment
- Vessels
- Ductwork systems
- Instrumentation

Piping Systems

Piping systems include piping to and from SC and SCN SSCs. 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 SSCs. Valves include the valve body assembly, actuators, appurtenances, and all non-electrical connections.

Rotating Equipment

Rotating equipment includes SC and SCN 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 SC and SCN heat exchangers and tanks, including their supports, filter assemblies, and nozzles.

Ductwork Systems

SC and SCN 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

SC and SCN 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:

- Missiles – A missile is an unrestrained mass with sufficient kinetic energy to cause damage to SC1 components. Definition of missile and missile protection requirements are addressed in Section 3A.3.2.7.
- 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 Section 3A.4.4.2.
- 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 Section 3A.4.4.2.

Fire Area and Fire Zone

A fire area is an area sufficiently bounded to withstand the hazards associated with the fire area and 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 bounded by fire-rated barriers.

Fire zone protection requirements are addressed in PSR Chapter 9B (Reference 3A-9). 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 Sections 3A.3.2.6 and 3A.4.2.1.

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

Section 3A.6.1.1 provides the operational transients, resulting loads, and load combinations.

Design loads and load combinations for fixed mechanical equipment are provided in Table 3A-11. Table 3A-15 through to Table 3A-24 present additional load combinations and address faulted level transients. 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 RPV, RPV internals, RPV supports, instrumentation, piping, electrical equipment, and the component supports.

For SC1, Safety Class 2 (SC2), and Safety Class 3 (SC3) containment structure penetrations, the anchor sleeve, where the sleeve itself acts as a containment boundary, is to meet the requirements of ASME BPVC, Section III, Subsection NE (Reference 3A-104).

Design for System Duty of Mechanical Systems Based on Frequency of Occurrence and Plant State

Table 3A-12 is used as a general event list for all hardware system duty design specifications. Service Levels are classified into service conditions as indicated below based on frequency of occurrence and plant state:

- Normal: Planned Operation
- Upset: AOO
- Emergency: DBA
- Faulted: DEC

The BWRX-300 utilizes the four service levels used in ASME BPVC Section III, Division 1 & 2, Subsection NCA, 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.

3A.6.1 Special Topics for Mechanical Components

This section addresses information concerning methods of analysis for BWRX-300 Seismic Category 1A and 1B components and supports, including both those designated as ASME BPVC, Section III, Division 1, Class 1, 2, 3, or Core Support and those not covered by the ASME BPVC as discussed in NUREG-0800 (Reference 3A-47) SRP 3.9.1. Information is presented concerning design transients for ASME BPVC Class 1 and Core Support components and supports.

The BWRX-300 design complies with the relevant requirements as stated below:

- The design meets the ASME BPVC, Section III, Division 1, Subsection NB acceptance criteria utilizing load combinations in Table 3A-11 including transient plant duty cycles in Table 3A-13 for the 60-year life period to assure minimal abnormal leakage. Rapidly propagating failure and gross rupture design is addressed in Section 3A.4.4.1.

- Mechanical components of the Reactor Coolant System (RCS) and associated auxiliary, control, and protection systems being designed with sufficient margin to ensure that the design conditions of the Reactor Coolant Pressure Boundary (RCPB) are not exceeded during any condition of normal operation, including AOOs. Sections 3A.6.1 through 3A.6.6 provide design requirements for SC and risk informed mechanical components that meet the margin and design condition requirements.
- Design quality control is discussed in PSR Chapter 17 (Reference 3A-26).
- Suitability of the plant design bases for mechanical components established in consideration of site seismic characteristics as discussed in Section 3A.3.1.4.

3A.6.1.1 Design Transients

The plant duty cycles represent transient events and their associated number of occurrences for the 60-year plant design life. These duty cycles are used for development of the BWRX-300 system and component design during normal operation, AOOs, DBAs, and DECs. Requirements are evaluated for the system design and performance as they relate to complete reactor operation. The duty is recorded as inputs to the system design for each specific primary and auxiliary hardware system. Duty is defined from a pressure and temperature perspective, mostly when variations in either variable are expected in important locations for the reactor including the RCPB and connecting piping and components.

The number of cycles associated with each event for the design of the RPV, RCPB, and other ASME pressure boundary components designed for fatigue are listed in Table 3A-13. Table 3A-12 breaks down the operational cycles by plant condition. The plant operating conditions are identified as Normal, AOO, DBA, DEC, or testing as defined in Section 3A.6.3.3. 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 3A-13 are documented in the design specifications and/or stress reports of the respective equipment. Table 3A-11 shows the load combinations and the standard acceptance criteria for ASME Section III components. Table 3A-15 through to Table 3A-24 provide the specific load combinations and acceptance criteria for ASME BPVC Section III, Division 1, Class 1, 2, and 3, B31.1 and B31.3 piping systems. Details about ASME B31.5 “Refrigeration Piping and Heat Transfer Component Code,” (Reference 3A-109), piping is addressed in detail in PSR Chapter 9A (Reference 3A-8).

3A.6.1.2 Computer Programs Used in Analyses

The computer programs used in the analyses of BWRX-300 Seismic Category 1A or 1B components are maintained by GEH or by outside computer program developers (see Appendix D).

The GEH software is controlled under NEDC-11209-A, “GE Hitachi Nuclear Energy Quality Assurance Program Description,” (Reference 3A-110). 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.

3A.6.1.3 Experimental Stress Analysis

Experimental stress analysis methods are used in compliance with the provisions of ASME BPVC, Section III (Reference 3A-105).

3A.6.1.4 Considerations for the Evaluation of Fault Conditions

Equipment designed to ASME BPVC, Section III, Division 1 is evaluated for the faulted (Service Level D) loading conditions. In these cases, the calculated actual stresses are compared to the allowable ASME BPVC, Section III, Division 1 Service Level D limits. The methods of analysis to calculate the stresses and deformations should conform to the methods outlined in the ASME BPVC, Section III, Division 1, “Mandatory Appendix XXVII, Design by Analysis for Service Level D,” (Reference 3A-111), subject to the conditions addressed in Subsection III of SRP 3.9.1. ASME QME-1, “Qualification of Active Mechanical Equipment Used in Nuclear Facilities,” (Reference 3A-112), as endorsed by Regulatory Guide (RG) 1.100, “Seismic Qualification of Electrical and Active Mechanical Equipment and Functional Qualification of Active mechanical Equipment for Nuclear Power Plants,” (Reference 3A-113), is used for functional qualification of valves.

Deformations under faulted conditions are evaluated in critical areas and the necessary design deformation limits, such as clearance limits, are satisfied.

The following sections address the evaluation methods and stress limits used for the equipment and identify the components evaluated for faulted conditions.

3A.6.1.4.1 Fine Motion Control Rod Drive

The Fine Motion Control Rod Drive (FMCRD) components that are part of the RCPB are analysed and evaluated for the ASME Service Level D faulted conditions in accordance with the ASME BPVC, Section III (Reference 3A-105). Refer to PSR Chapter 4 (Reference 3A-3) for FMCRD mechanism details.

3A.6.1.4.2 CRD Hydraulic Control Unit

The Hydraulic Control Unit (HCU) is analysed 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 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 PSR Chapter 4 (Reference 3A-3) for HCU description details.

3A.6.1.4.3 Reactor Pressure Vessel Assembly

The design of the RPV assembly as shown in PSR Chapter 4 (Reference 3A-3), out to and including the integral RIVs, RPV top head, housings for FMCRD, and in-core nuclear instrumentation complies with Subsections NB and NG of the ASME BPVC, Section III, Division 1 (Reference 3A-61) 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 I, ASME BPVC, Section III, Division 1, Subsection NB Class 1 valves. The ASME BPVC, Section III, Division 1 allowable stress is applied to assure integrity under applicable loading conditions including faulted conditions. The functional qualification for the RIV, includes analysis and/or testing for seismic and other dynamic conditions.

3A.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 CRD housings and other SC1 reactor internal components are evaluated for faulted conditions. The bases for determining the faulted loads for seismic events and other dynamic events are provided in PSR Chapter 3 (Reference 3A-1), Section 3.9. The allowable Service Level D limits for evaluation of these structures are per ASME BPVC, Section III, Division 1, Service Level D equations (Reference 3A-111) and Subsection NG.

For the shroud support, an elastic analysis is performed, and buckling is evaluated for compressive load cases for certain locations in the assembly.

3A.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 – "Rules for Construction of Nuclear Facility Components-Division 1, Subsection NF-Supports" (Reference 3A-97), 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 (RBV) events.

3A.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 I, ASME BPVC, Section III, Division 1, Subsection NB Class 1 and Subsection NCD Class 2 valves. The ASME BPVC, Section III, Division 1 allowable stresses are applied to assure integrity under applicable loading conditions including faulted conditions. The functional qualification of the active valves, including RIVs, CIVs, ICS, standby gas purge valves, and ICS condensate return valves includes analysis and/or testing for seismic and/or other dynamic conditions.

3A.6.1.4.7 Fuel Storage and Refuelling Equipment

The Fuel Storage and Fuel Handling equipment is described in detail in PSR Chapter 9A (Reference 3A-8). This includes the fuel pool structure, fuel racks, fuel pool cooling system, and fuel handling equipment.

Due to physical and structural separation, BWRX-300 Seismic Category 1A or 1B equipment is not 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 SSE without failure of the basic structure or damage to the active region of irradiated fuel.

3A.6.1.4.8 Fuel Assembly (Including Channel)

The fuel assembly (including channel) is described in detail in PSR Chapter 4 (Reference 3A-3).

The channel is subjected to mechanical tests to demonstrate the adequacy of the Global Nuclear Fuel (GNF) 2 channel for seismic/dynamic loads. The channel is tested to determine the allowable bending load that is 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 require that the SC1 components maintain the required functionality and structural capacity during ASME Level D service conditions.

3A.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, "Rules for Construction of Nuclear Facility Components – Division 1 – Subsection NCD-Class 2 and Class 3 Components," (Reference 3A-94).

3A.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 Class 3 pumps. The equivalent allowable stresses for pumps using elastic techniques are obtained from article NCD-3400 the ASME BPVC, Section III, Division 1, Subsection NCD (Reference 3A-94). The preliminary SSCs classification list does not identify any pumps to be ASME Class 2 or Class 3 currently.

3A.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 (Reference 3A-94).

3A.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 NB3600 (for Class 1 piping) of the ASME BPVC, Section III, Division 1, Subsection NB, Rules for Construction of Nuclear Facility Components – Division 1 – Subsection NB – Class 1 Components, (Reference 3A-93) and Article NCD-3600 (for Class 2 and 3 piping) of the ASME BPVC, Section III, Division 1, Subsection NCD (Reference 3A-94).

3A.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 is not reused. These two events are as follows:

- Postulated gross piping failure
- Postulated blow out of a CRD housing caused by a weld failure

The design criteria for pipe failure effects and mitigating features are provided in Section 3A.4.4. Except for the analysis of pipe failures, inelastic methods are not used in BWRX-300 piping design.

The mitigation of the CRD housing 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
- CRD outer tube
- Bayonet couplings

Only the bodies of the Control Rod Guide Tube, CRD housing, and CRD outer tube are analysed for energy absorption by inelastic deformation.

3A.6.2 Dynamic Testing and Analysis of Systems, Components and Equipment

This section 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 discussed in SRP 3.9.2. Structural requirements for conduits and cable tray supports and Heating, Ventilation, and Cooling System duct supports are specified in Section 3A.6.2.2.

The BWRX-300 design complies with the relevant requirements as stated below:

- Systems and components of the RCPB ae designed, fabricated, erected, and tested to have an extremely low probability of abnormal leakage, rapidly propagating failure or of gross rupture. The design meets the ASME BPVC, Section III, Division 1, Subsection NB acceptance criteria utilizing load combinations in Table 3A-11, including transient plant duty cycles in Table 3A-13 for the 60-year life period to assure minimal abnormal leakage. Abnormal leakage, rapidly propagating failure, and gross rupture design is addressed in Section 3A.4.4.
- RCS and associated auxiliary, control and protection systems are designed with sufficient margin to ensure that the RCPB is not breached during normal operating conditions, including AOOs of piping systems.
- Design quality control as discussed in PSR Chapter 17 (Reference 3A-26).
- Suitability of the plant design basis for mechanical components established in the consideration of site seismic characteristics as described in Section 3A.3.1.1.

3A.6.2.1 Piping Vibration, Thermal Expansion, and Dynamic Effects

Testing of piping systems involve the following systems:

- ASME BPVC, Section III (Reference 3A-105), Class 1, 2, and 3 piping systems
- High-energy piping systems inside BWRX-300 Seismic Category 1A structures or those whose failure would reduce function of a BWRX-300 Seismic Category 1A or 1B plant feature to an unacceptable level identified in Section 3A.4.4
- BWRX-300 Seismic Category 1A or 1B portions of moderate-energy piping systems located outside of containment

The overall test program is divided into two phases:

- Preoperational test phase
- Initial startup test phase

Piping vibration, thermal expansion, and dynamic effects testing is performed during both phases. Discussed below are the general requirements for this testing per ASME OM-2020 (Reference 3A-92), Nonmandatory Appendix M, Appendix M-3560 Division 2, Part 3 (Vibration Testing of Piping Systems) and Division 3, Part 7 (Thermal Expansion 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 sections that follow.

3A.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 are designed to withstand the dynamic effects of steady-state flow induced vibration and anticipated operational transient conditions.

3A.6.2.1.2 Thermal Expansion Testing

A thermal expansion preoperational and startup testing program verifies that normal unrestrained thermal movement occurs in specified high and moderate-energy piping systems. The testing is performed using visual observation and remote sensors. The purpose of this program is to ensure the following:

- The piping system during system heat up and cooldown is free to expand and move without unplanned obstruction or restraint in the x, y, and z directions
- The piping system does shake down after a few thermal expansion cycles

- The piping system is working in a manner consistent with the predictions of the stress analysis
- There is adequate agreement between calculated values and measured values of displacements
- There is consistency and repeatability in thermal displacements during heat-up and cooldown of the systems

The general requirements for thermal expansion testing of piping systems are specified in RG 1.68, “Initial Test Programs for Water-Cooled Nuclear Power Plants”. Detailed test specifications are prepared to address such issues as prerequisites, test conditions, precautions, measurement techniques, monitoring requirements, test hold points and acceptance criteria. The development and specification of the types of measurements required, the systems and locations to be monitored, the test acceptance criteria, and the corrective actions that may be necessary is provided prior to initial start-up.

In addition to thermal expansion testing, thermal stratification testing for the feedwater system piping is performed on the initial BWRX-300 plant. This testing is performed using external thermocouples on the pipe to confirm that the thermal stratification inputs to the piping analysis are conservative.

3A.6.2.2 Seismic Qualification of SC1 Mechanical Equipment (Including Other Reactor Building Vibration Induced Loads)

The following sections discuss the testing or analytical qualification of the SC1 mechanical equipment, and other ASME BPVC, Section III, Division 1 equipment including equipment supports.

3A.6.2.2.1 Tests and Analysis Criteria and Methods

PSR Chapter 3 (Reference 3A-1), Section 3.9 provides methodology for qualification of SC1 Mechanical Equipment.

3A.6.2.2.2 Qualification of Safety Class 1 Mechanical Equipment

CRD AND Control Rod Drive Housing (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.

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.

Control Rod Drive Housing Hydraulic Control Unit

The HCU is analysed 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.

Fuel Assembly (Including Channel)

The Fuel Assembly (including channel) qualification for seismic and faulted load conditions is described in PSR Chapter 4 (Reference 3A-3).

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 an SSE or other faulted RBV loadings is to close and remain closed after the event, except for ICS RIVs which are fail as-is type. This capability is demonstrated by the test and analysis.

Other ASME BPVC, Section III, Division 1 Structures, Systems and Components

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 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 of PSR Chapter 3 (Reference 3A-1).

Refer to Section 3.9 of PSR Chapter 3 (Reference 3A-1) for additional information on the dynamic qualification of valves.

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.

3A.6.2.3 Dynamic Response of Reactor Internals Under Operational Flow Transients and Steady-State Conditions

The reactor internal components within the vessel are subjected to extensive testing, coupled with dynamic system analyses, to evaluate the resulting flow induced vibration phenomena during normal reactor operation, and from anticipated operational transients. A report is developed of the complete flow induced vibration evaluation per the process outlined in USNRC RG 1.20, "Comprehensive Vibration Assessment Program for Reactor Internals During Preoperational and Startup Testing," (Reference 3A-114).

3A.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 (FIVs), 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.

3A.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.

3A.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
- Loss of Coolant Accident (LOCA) Loads
- Seismic Loads

3A.6.2.3.4 Correlations of Reactor Internals Vibration Tests with the Analytical Results

Prior to initiation of the instrumented vibration measurement program, 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 analysed 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 behaviour of the reactor internals. The additional knowledge gained from previous vibration tests is used in the generation of the dynamic models for seismic and LOCA analyses for the plant. The models used for the plant are similar to those used for the vibration analysis of earlier prototype Boiling Water Reactor (BWR) plants.

3A.6.3 ASME BPVC Class 1, 2, and 3 Components, Component Supports, and Core Support Structures

This section 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 ASME BPVC, Section III (Reference 3A-105), Division 1, Subsections NB, NCD, NF, and NG.

The BWRX-300 design complies with the relevant requirements as stated below:

- The RCPB is designed, fabricated, erected and tested, to have an extremely low probability of abnormal leakage, of rapidly propagating failure, and of gross rupture. The design meets the ASME BPVC, Section III, Division 1, Subsection NB (Reference 3A-93) acceptance criteria utilizing load combinations in Table 3A-11, including transient plant duty cycles in Table 3A-13, for the 60-year life period to assure minimal abnormal leakage. Rapidly propagating failure and gross rupture design is addressed in Section 3A.4.4.
- The RCS and its associated auxiliary, control and protection systems are designed with sufficient margin to assure that the design conditions of the RCPB are not exceeded during any condition of normal operation, including AOOs. The BWRX-300 auxiliary, control and protection systems associated with the RCS are designed on a graded quality approach and have sufficient redundancy to assure that the design conditions of the RCPB are not exceeded during and after any condition of normal operation, including AOOs. Sections 3A.6.1 through 3A.6.9 provide design requirements for SC and risk informed mechanical components that meet the margin, and design condition requirements.

3A.6.3.1 Loading Combinations, Design Transients, and Stress Limits

Section 7.3.3 delineates the criteria for selection and definition of design limits and loading combinations associated with Normal Operation, AOO, DBAs, DECs, and specified seismic and other RBV events for the design of ASME BPVC Section III, Division 1 components (except containment components which are discussed in Section 3A.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 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 Sections 3A.6.1 (Design Transients) and 3A.6.3 (Loading Criteria and Loading Phenomena). Seismic-related loads and methodology for seismic qualification are discussed in Section 3A.3.1.4. Table 3A-11 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, Subsection NG, "Rules for Construction of Nuclear Facility Components – Division 1 – Subsection NG - Core Support Structures," (Reference 3A-115). Specific loading combinations considered for evaluation of specific equipment are derived from Table 3A-11 and are contained in the design specifications and design reports for the respective equipment. For ASME Class 1 components where analysis for cyclic operation is evaluated in accordance with ASME BPVC, Section III, Division 1, Subparagraph NB-3222.4, the fatigue usage evaluation includes the use of environmental fatigue curves in accordance with RG 1.207, "Guidelines for Evaluating the Effects of Light-Water Reactor Water Environments in Fatigue Analyses of Metal Components," (Reference 3A-116) and NUREG/CR-6909, "Effect of LWR Water Environments on the Fatigue Life of Reactor Materials," (Reference 3A-117).

Specific load combinations and acceptance criteria for ASME Class 1 piping are shown in Table 3A-15. Also, for ASME 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 (70°F) 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 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.

3A.6.3.2 Design and Installation of Pressure Relief Devices

NEDC-33910P-A, Revision 2, "BWRX-300 Reactor Pressure Vessel Isolation and Overpressure Protection," (Reference 3A-118) describes the alternate methodology that is stated within ASME BPVC, Section III requirements for pressure relief. The BWRX-300 utilizes the ICS System for Pressure Relief as provided in PSR Chapter 6 (Reference 3A-5).

3A.6.3.3 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 PSR Chapter 15 (Reference 3A-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 service levels as shown in Table 3A-12.

The following are the plant condition events and transients associated with the BWRX-300 design:

3A.6.3.3.1 Normal Operation

See Table 3A-11 for the Load Combinations and acceptance criteria which includes normal operation conditions.

3A.6.3.3.2 Anticipated Operational Occurrences

See Table 3A-11 for the Load Combinations and acceptance criteria which includes AOO conditions.

3A.6.3.3.3 Design Basis Accident Events

See Table 3A-11 for the Load Combinations and acceptance criteria which includes DBA events.

3A.6.3.3.4 Design Extension Condition Events

See Table 3A-11 for the Load Combinations and acceptance criteria which includes DEC events.

3A.6.3.3.5 Seismic Events

Seismic design parameters and associated seismic events defined in Section 3A.3.1.2 are used in qualification of mechanical system components which considers non-LOCA faulted and plant testing events. PSR Chapter 3 (Reference 3A-1), Section 3.9 provides seismic qualification methodology to assure both component structural and/or functional capacity under seismic operational conditions are met.

All SSCs of the BWRX-300 design are designated by Safety Classification, quality group, and seismic category according to guidance in Section 3.2.

3A.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, Subsection NCA, “Rules for Construction of Nuclear Facility Components,” (Reference 3A-103).

SC1 electrical equipment is evaluated with respect to the load combinations in this document using International Electrotechnical Commission (IEC)/Institute of Electrical and Electronic Engineers (IEEE) 60780-323, “Nuclear Facilities – Electrical Equipment Important to Safety – Qualification,” (Reference 3A-119), and IEC/IEEE 60980-344 “Nuclear Facilities – Equipment Important to Safety – Seismic Qualification,” (Reference 3A-120).

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 QME-1, “Qualification of Active Mechanical Equipment Used in Nuclear Facilities” (Reference 3A-112).

3A.6.3.5 Acceptance Criteria

Components and supports comply with the design rules established for design, service, and test loadings in the appropriate section of ASME BPVC, Section III, Division 1 (Reference 3A-105).

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

3A.6.3.6 Loading Criteria

3A.6.3.6.1 Loading Conditions

The loadings that are considered in design a component include, but are not limited to, those below:

- Internal and external pressure
- Impact loads, including rapidly fluctuating pressures
- Weight of the component and normal contents under operating or test conditions, including additional pressure due to static and dynamic head of liquids
- Superimposed loads such as other components, operating equipment, insulation, corrosion resistant or erosion resistant linings, and piping
- Wind loads, snow loads, vibrations, and earthquake loads, where specified
- Reaction of supporting lugs, rings, saddles, or other types of supports
- Temperature effects

As appropriate, ASME BPVC, Division 1, Section III (Reference 3A-105), 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.

3A.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.

3A.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 to provide a complete basis for design, construction, and inspection.

For ASME BPVC, Section III, Division 1, Class 1 Components, the requirements below apply:

- Level B Conditions – The estimated duration of service conditions for which Level B limits are specified and included in the design specifications.
- 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 twenty-five stress cycles having an Sa value greater than that for 10^6 cycles from the applicable fatigue design curves of ASME BPVC 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, Section III, 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 ASME BPVC, Section III, Division 1, NCA-2142.1I.

3A.6.3.6.4 Test Loadings

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

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

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

3A.6.3.7 Loading Phenomena

This section describes the types of load phenomena, which are considered for components, as applicable. Consists of Pressure (P), Dead Weight (D), Fluid Reaction (R), prestress, fluid flow (including FIV when applicable) and other loads due to moving parts within a component or system.

The specified internal and external pressure (P) is the difference in pressure between the inside and outside of the item, or between any two chambers of a combination unit, which exists under the loadings for which the Level A Service Limits are applicable. Reactor Internal Pressure Differences, or ΔPs, are created across reactor internals as a result of fluid flow in the RPV. The service level differential pressure (ΔP) used in “N” is the maximum pressure expected to occur during each normal, upset, emergency and faulted service levels.

“N” loads in Table 3A-11 include Thermal Effects (T, defined next). Individual component specifications may elect to break out the thermal effects loads separately for the sake of clarity, if desired.

Thermal loads (TH) are included in this category for each service level, which are thermally induced loads caused by both steady state and transient operating conditions. The thermal anchor displacement loads (AN) are due to the anchor displacements (end motion) caused by thermal expansion, or pressure dilation, or other dynamic loads at anchor boundary locations.

3A.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 (NPP) systems, which produce vortex shedding (e.g., heat exchangers), pump (reciprocating or centrifugal), and thermodynamic instability conditions. Mechanical systems and equipment are reviewed for potential flow induced vibration mechanisms, evaluating all modes of system operation including both normal and abnormal conditions.

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

Consistently with SIL No. 682, “Flow Induced Vibration Load,” (Reference 3A-121), Flow Induced Vibration Load (alternative service level), the RPV component design considers the loads due to normal operation vibration (flow induced vibration) without combination of other primary loads.

Vortex Shedding

Vortex shedding occurs at certain fluid velocities when a fluid flows past objects. 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.

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), or the fluid flow forces acting on the valve parts during all modes of operations. Rapid valve closure or opening is defined as an AOO.

For example, Turbine Stop Valve (TSV) closure may be identified as being capable of generating large pressure waves which could cause significant dynamic response. Prior to TSV closure, saturated steam flows through MS 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 MS line through the vessel nozzle and into the vessel. This results in a compressive acoustic load and a steam impingement load on the steam dryer outer hood. Additionally, repeated reflections of the compression wave in the MS line generate time-varying forces in the MS piping. SSCs in the Reactor Building, Steam Tunnel and TB may be affected. TSV closure is conservatively evaluated as an AOO.

Failures of High-Energy Fluid System Piping

The effect of postulated pipe breaks in high-energy fluid systems as well as measures used to protect SSCs are defined in Section 3A.4.4. 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.

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 Section 3A.4.4.

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.

3A.6.3.8 Safety Class Equipment and Safety Category Functional Criteria

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

For any emergency or faulted service level, SC equipment and piping can accomplish their safety category functions as required by the event, but repairs could be required to ensure their ability to accomplish safety category functions as required by any subsequent design-condition event.

3A.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 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 stress analysis is performed on the RPV for various plant operating conditions (including faulted conditions) by using elastic methods, except as noted in Section 3A.6.1.4. Loading conditions, design stress limits, and methods of stress analysis for the core support structures, and other reactor internals are provided in Table 3A-11.

3A.6.3.10 Main Steam Piping

The MS piping trains extending from the outboard Main Steam Reactor Isolation Valve (MSRIV) to and including SIRs that are outboard of the Main Steam Containment Isolation Valves (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 ASME BPVC, Section III, Division 1. Table 3A-16 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, Section III, Division 1, Subsection NCD-3500 requirements for Class 2 components.

The MS system piping extending from the outboard SIR to the TSV is constructed in accordance with the ASME B31.1 criteria.

3A.6.3.11 Other Components

3A.6.3.11.1 Isolation Condenser System 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 1 requirements.

3A.6.3.11.2 Reactor Water Cleanup System Heat Exchangers

The Reactor Water Cleanup System (CUW) heat exchangers (regenerative) do not perform any Safety Category 1 functions. However, the heat exchangers are BWRX-300 Seismic Category Non-Seismic equipment. The ASME BPVC, Section VIII, Division 1, "Rules for Construction of Nuclear Facility Components – Appendices," (Reference 3A-122) requirements for SC3 components are used in the design and construction of the CUW heat exchanger components.

3A.6.3.11.3 Shutdown Cooling System Pump and Heat Exchangers

The Shutdown Cooling System (SDC) heat exchangers (nonregenerative) do not perform any Safety Category 1 functions. However, the pumps and heat exchangers are BWRX-300 Seismic Category Non-Seismic equipment, respectively. The ASME BPVC, Section VIII, Division 1 (Reference 3A-122) requirements for SC3 components are used in the design and construction of the SDC pump and heat exchanger components.

3A.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 ASME BPVC, Section III, Division 1. The analysis of these vessels is performed using elastic methods.

3A.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 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 Section 7.3.12 for additional information on valve operability.

3A.6.3.11.6 ASME BPVC, Section 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 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 ASME BPVC, Section III, Division 1. If an NB-3600 analysis is performed for ASME BPVC, Section III, Division 1, Class 2 or 3 piping, all analyses required for ASME BPVC, Section III, Division 1, Class 1 piping as specified in this document and the ASME BPVC are performed. Table 3A-15 and Table 3A-16 provide the specific load combinations and acceptance criteria for ASME BPVC, Section III, Division 1, Class 1, 2, and 3 piping systems.

3A.6.3.12 Valve Operability Assurance

This section discusses operability assurance of active ASME BPVC, Section III, Division 1 valves, including actuators (refer to PSR Chapter 3 (Reference 3A-1), Section 3.9 – Methods and Procedures of analysis or Testing of Supports for Mechanical, Electrical Equipment and Instrumentation).

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 Q20 or (for valve designs previously qualified to standards other than ASME QME-1), the following approach is used:

- Qualification specifications (e.g., design specifications) consistent with Appendices QVI and QV-A of QME-1 (Reference 3A-112) are prepared to ensure the operating conditions and Safety Category 1 functions for which the valves are to be qualified are communicated to the manufacturer or qualification facility.
- Suppliers are required to submit, for review and approval, application reports, as described in QME-1, which describe the basis for the application of specific predictive methods and/or qualification test data to a valve application.
- 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 deviations from QME-1 (Reference 3A-112) in the valve qualification. Each deviation is evaluated for effect 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 (Reference 3A-112).

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 (Reference 3A-112). For example:

- Evaluation of valve performance is based on a combination of testing and analysis, using design similarity to apply test results to specific valve designs
- Testing to verify proper valve setup and acceptable operating margin is performed using diagnostic equipment to measure stem thrust and torque, as appropriate
- 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
- Actuator sizing allows margin for aging/degradation, test equipment accuracy, and other uncertainties, as appropriate
- Material combinations that may be susceptible to galling or other damage mechanisms under certain conditions are not used

PSR Chapter 3 (Reference 3A-1), Section 3.9 provides details on the seismic qualification of valves and on the environmental qualification of values.

The SC1 SSCs meet the ASME BPVC, Section III, Division 1 requirements and perform their mechanical motion in conjunction with a dynamic (SSE or other RBV) load event. The dynamic qualification for operability is unique for each valve type; therefore, each method of qualification is provided individually below.

3A.6.3.13 Main Steam Containment Isolation Valves

The MSCIVs are SC1 and ASME Class 2 and are evaluated by analysis and testing for capability to operate under the design loads that envelop the predicted loads during a DBA and DBE. The IST testing requirements for MSCIVs are discussed in Section 3A.6.7.

3A.6.3.14 Other Active Valves

Other valves performing an active SC1 function 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 or other RBV event.

3A.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 valves to accomplish their intended functions, are developed to assure these functions are accomplished.

3A.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

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

3A.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

3A.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) as well as for complex assemblies such as control panels. It is reasonable to assume that a simple device, which 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 how the panel responds to 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.

3A.6.3.16 Design of Pressure Relief Devices

The RCS does not utilize safety or relief valves for overpressure relief. During normal operation, the MS flow to the turbine is throttled to control system pressure. PSR Chapter 6 (Reference 3A-5) describes the method of overpressure relief that is based on BWRX-300 Reactor Pressure Vessel Isolation and Overpressure Protection (Reference 3A-88).

3A.6.3.17 Component Supports

The establishment of the design/service loadings and limits is in accordance with ASME BPVC, Section III, Division 1, Article NCA 2000 and Subsection NF, “Rules for Construction of Nuclear Facility Components,” (Reference 3A-123). 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 stop, pipe whip restraints are not considered as pipe supports.

The design of bolts for component supports is specified in ASME BPVC, Section III, Division 1, Subsection NF (Reference 3A-123). 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.

3A.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 (Reference 3A-123). The building structure component supports (connecting the NF support boundary component to the existing building structure) are designed as specified in Sections 3A.3.2 (External Missiles) and 3A.4.3.

The design of supports for the piping satisfies the requirements of ASME B31.1 Power Piping Code, Paragraph 120 (Reference 3A-106).

3A.6.3.19 Reactor Pressure Vessel Stabilizer

The RPV stabilizer is designed as an SC1 linear type component support in accordance with the requirements of ASME BPVC, Section III, Division 1, Section NF (Reference 3A-123). 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 meet the ASME BPVC, Section III, Division 1 allowable stresses in the critical support areas for various plant operating conditions.

3A.6.3.20 Floor-Mounted Major Equipment

The condenser modules in the ICS are analysed to verify the adequacy of their support structures under various plant operating conditions. The analysis applies the maximum sheer, moment, and accelerations calculated from the seismic response analysis for the RB 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 stresses.

3A.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 section) are designed in accordance with ASME BPVC, Section III, Division 1, Subsection NF (Reference 3A-123) 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 3A-11. Active component supports are discussed in Section 3A.6.3.17. 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.

3A.6.4 Control Rod Drive System

The CRD System is described in PSR Chapter 4 (Reference 3A-3). The CRD System is an SC1 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 category functions that are performed. This provides an extremely high probability of accomplishing the safety category 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.

The BWRX-300 design complies with the relevant requirements below addressed as follows:

- The RCPB portion of the CRD system is designed, constructed, and tested for the extremely low probability of leakage or gross rupture. The design involves meeting the ASME BPVC, Section III, Division 1, Subsection NB (Reference 3A-93) acceptance criteria utilizing load combinations in Table 3A-11, including transient plant duty cycles in Table 3A-13 for the 60-year life period to assure minimal abnormal leakage. Rapidly propagating failure and gross rupture design is addressed in Section 3A.4.4.
- Reactivity control system redundancy and capability, as it relates to the CRD System, requires two independent reactivity control systems of different design principles to be provided. One of these systems is the control rods and the second is a reactivity control system that is capable of controlling reactivity under “planned, normal power changes.” One of the systems can hold the reactor core subcritical under cold conditions.
- The CRDS, in conjunction with reactor protection systems, is designed to assure an extremely high probability of accomplishing its safety functions in the event of AOOs. The design of the CRD System to the ASME BPVC, Section III, Division 1 Service Level A through D and test conditions as outlined in Table 3A-11, discussed in PSR Chapter 3 (Reference 3A-1), Section 3.9 for reactor protection system qualification, and discussed in PSR Chapter 4 (Reference 3A-3) assures accomplishment of safety functions in the event of an AOO.

3A.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 PSR Chapter 4 (Reference 3A-3).

3A.6.4.2 Applicable Control Rod Drive System Design Specification

The CRD System, which is designed to meet the functional design criteria outlined in PSR Chapter 4 (Reference 3A-3) consists of the following:

- Electro-hydraulic FMCRD mechanisms
- Hydraulic Control Unit
- Interconnecting Piping
- Instrumentation

The subcomponents of the FMCRD System forming part of the RCPB are designed according to ASME BPVC, Section III, Division 1, Subsection NB, Class 1 requirements (Reference 3A-93).

Pertinent aspects of the design and qualification of the CRD System components are discussed in the following locations: transients in Sections 3A.6.1.1, 3A.6.3.6 and 3A.6.3.7, faulted conditions in Section 3A.6.1.4, and seismic testing in PSR Chapter 4 (Reference 3A-3).

3A.6.4.3 Design Loads, Stress Limits, and 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 given in Table 3A-11.

3A.6.4.4 Control Rod Drive Performance Assurance Program

The assurance of the CRDM RCPB components to perform throughout the 60-year design life of the system is confirmed by the ASME Class 1 design report required by ASME BPVC, Section III, Subsection NB (Reference 3A-93).

Refer to PSR Chapter 4 (Reference 3A-3) for required tests to assure CRD operability.

3A.6.5 Reactor Pressure Vessel Internals

Refer to PSR Chapter 4 (Reference 3A-3) for a detailed description, design, and function.

The BWRX-300 design complies with the relevant requirements as stated below:

- Reactor internals are designed with appropriate margin to assure that specified acceptable fuel design limits are not exceeded during any condition of normal operation, including the effects of AOOs. The reactor internals consist of pressure boundary and, non-pressure boundary components, and support structures that are designed to maintain structural integrity per Sections 3A.6.1, 3A.6.2, and 3A.6.3, and maintain Specified Acceptable Fuel Design Limits as discussed in PSR Chapter 4 (Reference 3A-3).

3A.6.6 Functional Design, Qualification, and In-Service Testing Programs for Pumps, Valves and Dynamic Restraints

PSR Chapter 3 (Reference 3A-1), Section 3.9 provides the method for qualification of mechanical equipment. The qualification involves both determining component functionality while maintaining structural integrity under seismic, dynamic, and environmental conditions. Seismic testing of components is performed as well as use of analytical methods.

The BWRX-300 design complies with the relevant requirements as stated below:

- Designing, fabricating, erecting, and testing pumps, valves, and dynamic restraints that form the RCPB so as to have an extremely low probability of abnormal leakage, rapidly propagating failure, and gross rupture designed in accordance with Sections 3A.6.1, 3A.6.2, and 3A.6.3 for a 60-year life period to assure minimal abnormal leakage. Rapidly propagating failure and gross rupture design is addressed in Section 3A.4.4.

- Pumps, valves, and dynamic restraints that form the RCS and associated auxiliary, control, and protection systems are designed with sufficient margin to ensure that the design conditions of the RCPB are not exceeded during any condition of normal operation, including AOOs designed in accordance with Sections 3A.6.1, 3A.6.2, and 3A.6.3, and PSR Chapter 3 (Reference 3A-1), Section 3.9.
- Designing the Emergency Core Cooling System (ECCS) to permit periodic pressure and functional testing to ensure the leak tight integrity and operability and performance of its active components. The ECCS design incorporates adequate space and connections to allow for periodic pressure and functional testing.
- Designing periodic pressure and functional testing of the containment heat removal system to ensure the leak tight integrity and operability and performance of its active components. The containment heat removal system design incorporates adequate space and connections to allow for periodic pressure and functional testing.
- Designing the containment atmospheric cleanup systems to permit periodic pressure and functional testing to ensure the leak tight integrity and the operability and performance of the active components. The BWRX-300 containment atmosphere system is an inerted system design that incorporates adequate space and connections to allow for periodic pressure and functional testing.
- Designing the cooling water system to allow periodic pressure and functional testing to ensure the leak tight integrity and operability and performance of the active components. The cooling water system design incorporates adequate space and connections to allow for periodic pressure and functional testing.
- Designing piping systems penetrating containment with the capability to test periodically the operability of the isolation valves and associated apparatus and determine valve leakage acceptability. The piping system containment penetration design incorporates adequate space and connections to allow for periodic pressure and functional testing.
- Quality assurance in the design, fabrication, construction, testing, and records control of safety-related pumps, valves, and dynamic restraints.

IST Programs are developed for required operability and functional tests for components as described below:

- Necessary instrumentation, test connections, flow instruments, or other provisions that are required to fully comply with the requirements are included
- Valves required to be seat leakage tested have provisions for testing the leak-tightness of the valves in the direction in which the valves are required to be leak tight except as allowed in ASME OM-2020 (Reference 3A-92), ISTC-3630(b)
- ASME OM-2020 (Reference 3A-92), Nonmandatory Appendix M provides design guidance of system and component testing

The IST Program includes periodic tests and inspections that demonstrate the operational readiness of SC1 components, risk significant SC2 / SC3 pumps, valves, and snubbers (dynamic restraints) and their capability to perform their safety category functions. Table 3A-14 provides a preliminary list of active valves. The IST Program is based on the requirements of ASME OM-2020 (Reference 3A-92) and ASME Section XI, “Rules for Inservice Inspection of Nuclear Power Plant Components,” (Reference 3A-124) as follows:

- The specific ASME OM Code requirements for functional testing of pumps are found in Subsection ISTF for plants designed after 2000
- Requirements for IST of valves are found in Subsection ISTC and Mandatory Appendices III (motor-operated valves) and IV (air-operated valves)
- Requirements for dynamic restraints (snubbers) are found in Subsection ISTD
- Requirements for IST of pressure relief devices are found in Mandatory Appendix I
- Requirements for Check Valve Condition Monitoring are found in Mandatory Appendix II
- General requirements for IST are found in Subsection ISTA

3A.6.6.1 IST of Pumps

The BWRX-300 design does not require the use of pumps to perform a specific function in shutting down the reactor to the safe shutdown condition, maintaining the safe shutdown condition, or mitigating the consequences of an accident. Therefore, there are no pumps required to be included in the IST Program.

3A.6.6.2 IST of Valves

Certain ASME BPVC Classes 1, 2, and 3 valves and pressure relief devices are subject to IST in accordance with ASME OM-2020 (Reference 3A-92), Subsection ISTC or Appendix I, including the general requirements in Subsection ISTA. IST of valves assesses operational readiness, including actuating and position-indicating systems. The valves that are subject to IST include those valves that perform a specific function in shutting down the reactor to a safe shutdown condition, maintaining a safe shutdown condition, or mitigating the consequences of an accident. In addition, pressure relief devices used for protecting systems or portions of systems 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, are subject to IST.

ASME OM-2020 (Reference 3A-92), Table ISTC-3500-1, requires the following four basic valve tests:

- Exercise tests
- Seat leakage tests
- Remote position indicator tests
- Special tests (e.g., fail-safe tests, explosive valve tests, rupture disk tests)

3A.6.6.3 Valve Categories

Non-exempt ASME BPVC Class 1, 2, and 3 valves are categorised in accordance with ASME OM-2020 (Reference 3A-92), ISTC-1300 (2) as follows:

- Category A – valves for which seat leakage is limited to a specific maximum amount in the closed position for fulfilment of their required function(s)
- Category B – valves for which seat leakage in the closed position is inconsequential for fulfilment of the required function(s)
- Category C – valves that are self-actuating in response to some system characteristic, such as pressure (relief valve) or flow direction (check valve) for fulfilment of the required function(s)
- Category D – valves that are actuated by an energy source capable of only one operation, such as rupture disks or explosively actuated valves

When more than one distinguishing category characteristic is applicable, all requirements of each of the individual categories are applicable, although duplication or repetition of common testing requirements is not necessary.

3A.6.6.4 Valve Exercise Tests

Active Category A valves, Category B valves, and Category C check valves are exercised periodically, except for self-actuated safety and relief valves. The ASME OM Code specifies a quarterly valve exercise frequency for all valves except power-operated safety and relief valves, which are required to be tested once per fuel cycle. Manual valve exercise tests are discussed in ASME OM-2020 (Reference 3A-92), ISTC-3540. Where it is not practicable to exercise a valve during normal power operation, the valve exercise test is deferred to either cold shutdown or refuelling outages. In some cases, quarterly stroke testing is deferred to refuelling outages or cold shutdown. The bases for deferral are consistent with ASME OM-2020, ISTC-3500. Where practical, the BWRX-300 is designed to accommodate quarterly stroke testing.

During valve exercise tests, the necessary valve obturator movement is determined by exercising the valve while observing an appropriate direct indicator, such as indicating lights that signal the required changes of obturator position, or by observing other evidence or positive means, such as changes in system pressure, flow, level, or temperature that reflect change of obturator position.

Check valve exercise tests use direct observation or other positive means (ASME OM-2020, ISTC-5221(a)) for verification of valve obturator position by performing both an open and a close test.

Active and passive Category A containment isolation valves are tested to verify seat leakage is within limits.

Other Category A valves are required to be seat leakage tested at least once every two years as specified by ASME OM-2020 (Reference 3A-92), ISTC-3630.

Check valves that have seat leakage requirement are leak tested in accordance with ASME OM-2020 (Reference 3A-92), ISTC-3600.

The USNRC allows licensees to apply the ASME OM Code Cases listed in USNRC Regulatory Guide 1.192, “Operation and Maintenance Code Case Acceptability, ASME OM Code,” (Reference 3A-125), as incorporated by reference in paragraph 10 CFR 50.55a (a)(3)(iii), without prior USNRC approval, subject to the conditions in 10 CFR 50.55a(b)(6) (Reference 3A-38).

Relief from the testing requirements of the ASME OM Code is requested when compliance with requirements of the ASME OM Code is not practical. In such cases, specific information is provided which identifies the impractical code requirement, justification for the relief request, and the testing method to be used as an alternative. Demonstration of the impracticality of the testing required by the ASME OM Code, and justification for alternative testing proposed, are provided.

The BWRX-300 IST Program does not invoke the use of any ASME code cases for valve IST.

3A.6.6.5 IST of Snubbers

- At this time, no snubbers are included in the BWRX-300 design. This could change as civil design matures. Section 3A.6.2.2 discusses methodology for qualification of dynamic restraints.
- Preservice operability testing is to be performed on all snubbers. Tests verify the parameters described in ASME OM-2020 (Reference 3A-92), ISTD-5120.

- Snubbers are tested at a load sufficient to verify the test parameters specified in ASME OM-2020 (Reference 3A-92), ISTD-5120. Any snubbers that fail the preservice operational readiness test are evaluated for causes of failure. If a design deficiency is found, it is corrected, adjusted, repaired, or replaced. Failed snubbers are tested to meet the requirements in ASME OM-2020 (Reference 3A-92), ISTD-5120.
- Snubber operational readiness checks verify the parameters described in ASME OM-2020 (Reference 3A-92), ISTD-5120.
- Snubbers are tested at a load sufficient to verify the test parameters specified in ASME OM-2020 (Reference 3A-92), ISTD-5100, ISTD-5200, and ISTD-5300, or ISTD-5400 and ISTD-5500.
- Snubbers are tested for operational readiness during each fuel cycle as defined in ASME OM-2020 (Reference 3A-92), ISTD-5240. Test campaigns are required to be in accordance with a specified sampling plan as defined in ASME OM-2020 (Reference 3A-92), ISTD-5260.

The snubbers IST sample is selected using one of the following two sample plans:

- 10% testing sample plan
- 37 testing sample plan (test 37 snubbers)

For more detailed requirements on testing sample plans, refer to ASME OM-2020 (Reference 3A-92), ISTD-5300 and ISTD-5400.

- Snubbers preservice and IST, including their interval and percent of sample lot size requirements, are performed in accordance with ASME OM-2020(Reference 3A-92), ISTD.
- Snubbers testing documentation necessary to verify the results of the preservice and IST Program is in accordance with the requirements in ASME OM-2020 (Reference 3A-92), ISTD 9000.

3A.6.7 Risk Informed In-Service Testing

Based on system and component performance, the licensee will assess application of this to BWRX-300 later in operational life.

3A.6.8 Risk Informed Inservice Inspection of Piping

Based on system and component performance, the licensee will assess application of this to BWRX-300 later in operation life.

3A.6.9 ASME Code Class 1, 2, and 3 Piping Systems, Piping Components and Their Associated Supports

This section provides the general design aspects used for SC and SCN piping systems, piping components and pipe supports. The design of SC piping systems, piping components and pipe supports is based on the code rules established under the ASME BPVC, Section III, Division 1 for Class 1, Class 2, and Class 3 nuclear piping, components and supports. For non-Division 1 components, ASME B31.1 power piping (Reference 3A-106), ASME B31.3 process piping (Reference 3A-107), and ASME B31.5 (Reference 3A-109) piping codes are used. 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.

The load combinations and acceptance criteria in Table 3A-15 are applied to the analysis of Class 1 piping including its interfaces with components and penetrations.

The load combinations and acceptance criteria in Table 3A-16 are applied to the analysis of Class 2 and 3 piping systems and containment penetrations.

The load combinations and acceptance criteria in Table 3A-17 are applied to the analysis of nonnuclear (ASME B31.1) piping systems and components.

The load combinations and acceptance criteria in Table 3A-18 are applied to the analysis of non-nuclear (ASME B31.3) piping systems and components.

The load combinations and acceptance criteria in Table 3A-19 are applied to the analysis of snubber type supports.

The load combinations and acceptance criteria in Table 3A-20 are applied to the analysis of rigid type supports.

The load combinations and acceptance criteria in Table 3A-21 are applied to the analysis of ASME Class 1 piping connected to RPV nozzles.

The load combinations and acceptance criteria in Table 3A-22 are applied to the analysis of ASME Class 2 and 3 piping connected to RPV nozzles.

The load combinations and acceptance criteria in Table 3A-23 are applied to the analysis of pipe mounted valves.

The load combinations and acceptance criteria in Table 3A-24 are applied to the analysis of ASME B31.1 and ASME B31.3 Linear Type Supports (Anchor, Guide, Struts).

The BWRX-300 meets the relevant requirements:

- Design transients and resulting load combinations for SC1 piping and pipe supports have been developed as necessary to withstand the effects of earthquakes combined with the effects of normal or accident conditions including a LOCA and dynamic effects as discussed in Section 4.1.3 and as defined in Table 3A-15, Table 3A-16, Table 3A-17 and Table 3A-18.
- The RCPB of the piping designs are designed, fabricated, erected, and tested to have an extremely low probability of abnormal leakage, of rapidly propagating failure, and of gross rupture. The design involves meeting the ASME BPVC, Section III, Division 1, Subsection NB (Reference 3A-93) acceptance criteria utilizing load combinations in Table 3A-15; including transient plant duty cycles in Table 3A-13 for the 60-year life period to assure that minimal abnormal leakage occurs. Rapidly propagating failure and gross rupture design are addressed in Section 3A.4.4.
- The RCS and associated auxiliary, control and protection systems are designed with sufficient margin to ensure that the design condition of the RCPB are not exceeded during any condition of normal operation, including AOOs. Sections 3A.6.1 through 3A.6.6 provide design requirements for SC and risk informed mechanical components and piping that meet the margin and design condition requirements.

3A.6.9.1 Piping Analysis Methods

ASME Class 1 piping design conforms to the requirements of ASME BPVC, Section III, Division 1, Subsection NB for 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, Subsection NF.

ASME Class 2/3 piping design conforms to the requirements of ASME BPVC, Section III, Division 1, Subsection NCD code rules that covers both piping and piping components. The pipe supports attached to the ASME Class 2 and 3 piping meets the appropriate requirements of ASME BPVC Section III, Division 1, Subsection NF.

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. Containment mechanical penetrations and penetration design details are discussed in PSR Chapter 6 (Reference 3A-5).

Power piping is designed per ASME B31.1, “Power Piping”, and process piping is designed per ASME B31.3, “Process Piping”. Descriptions of systems that contain ASME B31.1 or ASME B31.3 piping and components including their functions are described in the system chapters.

Refrigerant piping meets the requirements of ASME B31.5, “Refrigeration Piping and Heat Transfer Components” (Reference 3A-109). Non-refrigerant piping is designed in accordance with the requirements of ASME B31.1, with the exception of the containment penetration portion. The CIVs and lines that supply and return chilled water to the containment cooling system cooling coils are described in PSR Chapter 6 (Reference 3A-5).

Table 3A-15 and Table 3A-16 provide the specific load combinations and acceptance criteria for ASME BPVC, Section III, Division 1, Class 1, 2, and 3 piping including its interfaces with components and penetrations. Table 3A-17 and Table 3A-18 provide the specific load combinations and acceptance criteria for ASME B31.1 and ASME B31.3 code piping.

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 Subsection NF support boundary component to the existing building structure) are designed as specified in Section 3A.5.2.2.

3A.6.9.1.1 Experimental Stress Analysis Methods

Standard flanges used in safety-related piping systems are in accordance with ASME B16.5, “Pipe Flanges and Flanged Fittings; NPS ½ through NPS 24, metric/Inch Standard,” (Reference 3A-126), “Pipe Flanges and Flanged Fittings: NPS 1/2 Through NPS 24, Metric/Inch Standard” with high-strength bolting per NCD-3658.3.

In cases where a fitting or joint design is used which is not covered by ASME BPVC, Section III, Division 1, Subparagraph Table NB-3681(a)-1, (or if it is desired to determine less conservative values), stress intensification factors and flexibility factors is established using experimental or analytical data.

Determination of non-standard stress intensification factors and flexibility factors are in accordance with the rules of ASME BPVC, Section III, Division 1, Subparagraph NB-3672.7 (Reference 3A-61).

3A.6.9.1.2 Modal Response Spectrum Method

The seismic response of the piping system is determined by performing a modal analysis by either the Response Spectrum Method or Time History Method. The procedure for multi-support excitation is followed with both methods.

The spectral peak shifting procedures of ASME BPVC, Section III, Subsection Appendices, Nonmandatory Appendix N, “Rules for Construction of Nuclear Facility Components,” (Reference 3A-127), are used for design of systems with closely spaced structural modes and artificially broadened spectra input.

Nodal points are selected to coincide with the locations of concentrated masses, locations of significant geometry changes, or locations of supports. Node points need to be sufficiently close to ensure higher modes to be captured.

The support movement of multiple-supported piping and equipment systems subject to seismic and other dynamic load excitations are considered in design verification. As a conservative approach, the maximum displacement of each support point can be computed by either time history analysis, or response spectrum analysis. The computed individual maximum support displacements are then imposed on the supported piping and equipment systems in the most limiting, or severe, directional combination for the piping and equipment design verification consideration.

Refer to Section 3A.3.1.4 and PSR Chapter 3 (Reference 3A-1), Section 3.9 for discussions on model response methods of piping.

3A.6.9.1.3 Response Spectra Method – Independent Support Motion Method

Independent Support Motion or Multiple Response Spectra Method is used when piping systems are supported by multiple support structures or at multiple levels within a structure. In this method of analysis, supports are divided into support groups with different seismic excitation applied to each group. A single set of response spectra is applied to all supports of each group, but different set of response spectra are applied to different groups. Typically, a support group is made up of supports attached to the same structure, floor, or portion of a floor.

In the Independent Support Motion Method, dynamic analysis is performed by applying the appropriate response spectra at one support group at a time, with no motion applied to the other supports. The process is repeated for each support group.

The combinations of modal responses and spatial components for systems analysed using Independent Support Motion is performed consistent with NUREG-1061, “Evaluation of Other Loads and Load Combinations,” (Reference 3A-128). Inertial responses from multiple groups in the same direction are combined by the Absolute Summation Method. Modal and directional responses are combined by SRSS method without considering closely spaced frequencies. The responses due to relative displacements at the support points are combined with the inertial responses by the SRSS method.

If the seismic inputs are known to be in-phase, the combination of responses are accomplished by the Absolute Summation Method.

See Section 3A.3.1.4 for the Seismic Analysis of Piping Systems.

3A.6.9.1.4 Time History Method

Section 3A.3.1.3 describes the Three Components of Design Ground Motion analyses which conforms to the acceptance criteria described in Regulatory Guide (RG) 1.92 (Reference 3A-60). The time history modal superposition, time history direct integration, and response spectrum modal superposition methods can all be used in Multi-Support Excitation analysis. The modal and spatial combinations in the time history method are described in RG 1.92 (Reference 3A-60).

3A.6.9.1.5 Inelastic Analysis Method

Piping system stresses are calculated on an elastic basis for each service level and evaluated in accordance with ASME BPVC, Section III, Division 1, Subarticles NB-3600 (ASME 1) and NCD3600 (ASME 2&3) as described in Section 3A.6.1.4. Inelastic analysis methods are addressed in Section 3A.6.1.4. Mathematical models for piping systems reflect the static and dynamic characteristics of the system. In general, a piping system consists of pipe elements such as straight sections, fittings, bends, and valves. The piping system is supported and restrained by hangers, anchors, pipe guides, struts, box frames, snubbers, and equipment nozzles.

Large equipment, such as vessels and pumps, is typically decoupled from piping analysis at the nozzle connection.

The mass of the pipe and inline equipment is lumped at corresponding nodal locations and connected by weightless elastic beam elements to reflect the physical properties of the corresponding piping segment.

Inelastic analysis method is not used in the BWRX-300 piping design and analysis except for piping whip restraints in Section 3A.4.4. For Class 1 piping of which shakedown is not established by meeting Equation (10) of NB-3600, the simplified elastic plastic discontinuity analysis is performed through Equation (12) to Equation (14) of NB3600 to limit the accumulated plastic strains. The system piping analysis model includes the valves and nozzle connections to pumps and other large equipment. Level D loads acting on the components are determined through the system piping analysis.

3A.6.9.1.6 Non-Seismic and Seismic Category 2

Section 3A.3.1.4 describes the design of the interaction of the BWRX-300 SSCs that are categorised as BWRX-300 non-seismic/Seismic Category 2 with SSCs that are categorised as Seismic Category 1A or 1B.

In certain instances, BWRX-300 Seismic Category 2 piping is connected to BWRX-300 Seismic Category 1A or 1B piping at locations other than a piece of equipment which, for purposes of analysis, could be represented as an anchor. The transition points typically occur at BWRX-300 Seismic Category 1A or 1B valves, which may or may not be physically anchored. Because a dynamic analysis is modelled from pipe anchor point to anchor point, two options exist:

- Specify and design a structural anchor at the BWRX-300 Seismic Category 1A or 1B valve and analyse the BWRX-300 Seismic Category 1A or 1B subsystem.
- Analyse the subsystem from the anchor point in the BWRX-300 Seismic Category 1A or 1B subsystem through the valve to either the first anchor point in the BWRX-300 Seismic Category 2 subsystem, or for a distance such that there are at least two seismic restraints in each of the three orthogonal directions.

For either option, seismic loads are applied on all anchors and rigid supports in the model.

Where small BWRX-300 Seismic Category 2 piping is directly attached to BWRX-300 Seismic Category 1A or 1B piping, it may be decoupled from BWRX-300 Seismic Category 1A or 1B piping following guidelines in Section 3A.6.9.2.

3A.6.9.1.7 Small Bore Piping Method

Small bore piping is defined as piping 50 mm and less nominal pipe size. Decoupling of small branch piping from main run piping follows decoupling criteria in Section 3A.6.9.2. Seismic analysis of small-bore piping follows dynamic analysis methods per Section 3A.6.9.1, or the equivalent static coefficient method if justified.

3A.6.9.1.8 Category I Buried Piping

The BWRX-300 design does not include ASME Code Class 1, Seismic Category I buried piping.

3A.6.9.2 Piping Modelling Techniques

3A.6.9.2.1 Computer Codes

The computer programs used in the mechanical system and component analyses of the SC1 components are described in Appendix E and Section 3A.6.1.2.

3A.6.9.2.2 Dynamic Piping Model

Sections 3A.6.3 and 3A.3.1.4 describe the analyses that are used to confirm that the BWRX-300 Seismic Category 1A or 1B subsystems, including piping and supports, function during and after an earthquake.

3A.6.9.2.3 Piping Benchmark Program

Refer to Appendix E for discussion of how the computer codes are benchmarked with the appropriate USNRC benchmarks in accordance with NEDC-11209-A, "GE Hitachi Energy Quality Assurance Program Description" (Reference 3A-110).

3A.6.9.2.4 Decoupling Criteria

Decoupling is a method of dividing or separating analysis models based on pipe size/moment of inertia. Branch lines with a run to branch moment of inertia ratio of 25 to 1 or greater are decoupled. The decoupling criteria described in Section 3A.3.1.3 includes BWRX-300 Seismic Category 1A or 1B subsystems which includes piping and supports. Piping design specifications use an alternative approach stated in Welding Research Council Bulletin WRC-300 Part II, "Technical Position on Damping and on Industry Practice," (Reference 3A-129) to decouple the subsystem. This approach aligns with the standard industrial practice.

3A.6.9.3 Piping Stress Analysis Criteria

Piping stress criteria for ASME BPVC, Section III, Division 1, Class 1, 2, and 3 piping are discussed in Section 7, Table 3A-15 and Table 3A-16.

3A.6.9.3.1 Seismic Input

Seismic Input criteria for In-Structure Response Spectra for ASME BPVC, Section III, Division 1, Class 1, 2, and 3 piping are discussed in Section 4.1.2.

3A.6.9.3.2 Design Transients

Design transients and dynamic loading for ASME BPVC, Section III, Division 1, Class 1, 2, and 3 piping are covered in Sections 3A.6.1 (Design Transients) and 3A.6.3 (Loading Combinations, Design Transients, and Stress Limits).

3A.6.9.3.3 Loadings and Load Combinations

Table 3A-15, Table 3A-16, Table 3A-17 and Table 3A-18 provide the specific load combinations and acceptance criteria for ASME BPVC, Section III, Division 1, Class 1, 2, and 3 and non-ASME class piping systems, structures, and components.

3A.6.9.3.4 Damping Ratios

Damping values for seismically qualified ASME piping are discussed in Sections 3A.3.1.2 and 3A.3.1.4, and provided in Table 3A-4 for seismic damping values for piping, equipment, and equipment supports.

Floor response spectra include 15% peak broadening to account for possible variations in structural frequencies.

Floor response spectra used for the qualification of Class 1, 2, and 3 piping for dynamic loads with Level B service limits use a 3% critical damping ratio for piping greater than or equal to 300 mm NPS, and a 2% critical damping ratio for piping less than 300 mm NPS.

Floor response spectra used for the qualification of Class 1, 2, and 3 piping for dynamic loads with Level C or Level D service limits use a 4% critical damping ratio for all pipe sizes.

3A.6.9.3.5 Combination of Modal Responses

The modal and spatial combinations in the time history method are described in RG 1.92 (Reference 3A-60).

3A.6.9.3.6 High-Frequency Modes

Cutoff frequency and effect of missing mass to be considered for dynamic analysis are determined in accordance with Section 3A.3.1.4.

The effect of the missing mass associated with the cutoff frequency is properly considered in the design of the system and the support of the structures and components.

The additional constraint forces associated with the missing mass are obtained by applying an additional force at the centre of mass of the equipment. This force is conservatively calculated as the product of the total missing mass and the maximum input acceleration spectrum values beyond the cutoff frequency. A less conservative method may be used when justified.

3A.6.9.3.7 Fatigue Evaluation for ASME Code Class 1 Piping

The Fatigue Evaluation for ASME Code Class 1 Piping conforms to ASME BPVC, Section III, Division 1, Subsection NB (Reference 3A-93) requirements as described in Section 3A.6.9.3 and Table 3A-15 which includes the fatigue usage and stress limits and conforms to ASME BPVC, Section III, Division 1, Paragraph NB-3653. The Cumulative Usage Factor (CUF) is calculated in accordance with ASME BPVC, Section III, Division 1, Subparagraph NB-3653.5 and RG 1.207 (Reference 3A-116).

The number of earthquake events of SSCs for seismic fatigue effects is based on the dynamic response resulting from the occurrence of only one SSE during the operating lifetime of the NPP.

3A.6.9.3.8 Fatigue Evaluation of ASME Code Class 2 and 3 Piping

The Fatigue Evaluation for ASME Code Class 2 and 3 Piping conforms to ASME BPVC, Section III, Division 1, Subsection NCD requirements (Reference 3A-94).

In cases where a fitting or joint design is used which is not covered by ASME BPVC, Section III, Division 1, Subparagraph Table NCD3673.2(b)-1 (or if it is desired to determine less conservative values), stress intensification factors and flexibility factors are established using experimental or analytical data.

Determination of non-standard stress intensification factors and flexibility factors is in accordance with the rules of ASME BPVC, Section III, Division 1, Subparagraph NCD-3673.2.

Expansion and flexible hoses are designed in accordance with the requirements of ASME BPVC, Section III, Division 1, Paragraph NCD-3649. Expansion joints and flexible hoses are subject to limitations in displacement during operation. The component manufacturer provides these displacement limits. The displacement values obtained from the stress analysis are reviewed to ensure that the specified displacement limits have not been exceeded.

3A.6.9.3.9 Thermal Oscillations in Piping Connected to the Reactor Coolant System

Thermal stratification testing of the feedwater piping is discussed in Section 3A.6.2.1.

As described in USNRC Bulletin 88-08, “Thermal Stresses in Piping Connected to Reactor Coolant Systems,” (Reference 3A-130), thermal fatigue of unisolable piping connected to the RCS can occur when the connected piping is isolated by a leaking block valve, the pressure upstream from the block valve is higher than RCS pressure, and the temperature upstream is significantly cooler than RCS temperature. An unrecognized phenomenon and possibly unanalysed condition may exist for those reactors that are subjected to these conditions. Under these conditions, thermal fatigue of the unisolable piping can result in crack initiation. Unisolable sections of piping connected to the RCS which may be subjected to the temperature stratification/oscillation is identified where such conditions are analysed.

3A.6.9.3.10 Thermal Stratification

Transient loading due to hot and cold fluid not mixing at low flow conditions is considered in the design of the piping. Stresses due to the effects of thermal stratification are evaluated using ASME BPVC, Section III, Rules for Construction of Nuclear Facility Components – Appendices, Non-Mandatory Appendix J. 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 considered and properly included in the thermal expansion analysis. Thermal stratification testing of the feedwater piping is discussed in Section 3A.6.2.1.

The stresses and fatigue associated with thermal stratification and thermal stripping are considered in the piping analyses that demonstrate compliance within ASME Class 2 and 3 code limits. The global effect of thermal stratification causes the bending of the piping system due to the temperature distribution in the element cross section. Consequently, this generates extra thermal stress in the load combination as defined in Table 3A-17.

3A.6.9.3.11 Safety Relief Valve Design, Installation and Testing

No ASME Code Class 1, 2, or 3 safety relief valves are in the BWRX-300 design. See PSR Chapter 6 (Refence 3A-5) Sections 6.2 and 6.8.3 which describes the installation of BWRX-300 overpressure protection pressure relief devices.

If ASME Code Class 1 safety relief valves are required, design follows the criteria for pressure relief installations specified in ASME BPVC, Section III, Non-mandatory Appendix O. In addition, the following criteria are evaluated:

- Where more than one valve is installed on the same pipe run, the sequence of valve openings to be assumed in analysing for the stress at any piping location is that sequence which is estimated to induce the maximum instantaneous value of stress at that location.
- Stresses are evaluated, and applicable stress limits are satisfied for all components of the pipe run and connecting systems and the pressure relief valve station, including supports and all connecting welds between these components.
- In meeting the stress limit requirements, the contribution from the reaction force and the moments resulting from that force include the effects of a Dynamic Load Factor (DLF) or use the maximum instantaneous values of forces and moments for that location as determined by the dynamic hydraulic/structural system analysis. This requirement is satisfied in demonstrating satisfaction of all design limits at all locations of the pipe run and the pressure relief valve for Class 1 piping.

In the case that a Class 1 piping system is designed without a safety relief valve, the overpressure protection design follows ASME BPVC, Section III, Division 1, Subparagraph NB-7120(c).

If ASME Code Class 2 or 3 safety relief valves are required, the design follows the criteria for pressure relief installations specified in ASME BPVC, Section III, Nonmandatory Appendix O.

In the case that a Class 2 or 3 piping system is designed without a safety relief valve, the overpressure protection design follows ASME BPVC, Section III, Division 1, Subparagraph NCD7120(c).

3A.6.9.3.12 Functional Capability

Functional capability for ASME Service Level D load conditions is discussed in Section 3A.6.1.4 and is applicable to SC1 SSCs.

If the requirements of NUREG-1367, "Functional Capability of Piping Systems," (Reference 3A-131) are applicable, functional capability of ASME Class 1, 2, and 3 piping systems essential for the safe shutdown under postulated events is ensured provided the following requirements are met:

- The conditions of the equation in ASME BPVC, Section III, Division 1, Paragraph NB, NB3652, Equation (9) are met for Service Level C and D loads, (Class 1)
- The conditions of the equation in ASME BPVC, Section III, Division 1, Subparagraph NCD-3653.1(a)(9a) are met for Service Level C and D loads, (Class 2 and 3)

- Dynamic moments are calculated using an elastic response spectrum analysis with +/- 15% peak broadening and with no more than 5% damping
- Steady state (weight) stress is less than or equal to 25% of yield stress
- The ratio of pipe diameter to wall thickness is less than or equal to 50
- External pressure is less than or equal to internal pressure

3A.6.9.3.13 Combination of Inertial and Seismic Anchor Motion Effects

Combination of inertial and seismic anchor motion effects are discussed in Section 3A.6.3.6.

3A.6.9.3.14 Operating Basis Earthquake as a Design Load

See PSR Chapter 15.8 (Reference 3A-15) for further details on consideration of the Operating Basis Earthquake (OBE). Design load combinations do not consider OBE and site operating earthquake loads, 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 Section 3A.3.1.2.

3A.6.9.3.15 Welded Attachments

The piping system and pipe supports are designed and located to minimize the necessity for welded attachments to the pipe. Where welded attachments exist, local stress evaluations are performed.

The code requirements are met with the local stress added to the regular piping stress.

The local stresses are evaluated per ASME BPVC, Section III Appendices (Reference 3A-122), Nonmandatory Appendix Y.

Use of stiff pipe clamps may result in local pipe stresses. Stiff pipe clamps described in Issue 89 of NUREG-0933, Resolution of Generic Safety Issues are not to be used.

3A.6.9.3.16 Modal Damping for Composite Structures

Modal Damping for Composite Structures is defined in Section 3A.3.1.2.

Cross-reference with the following Tables in Section 3A.3.1:

- Table 3A-2: Seismic Damping Values for BWRX-300 Structures.
- Table 3A-3: Seismic Damping Values for RPV and Internals.
- Table 3A-4: Seismic Damping Values for Piping and Equipment.

3A.6.9.3.17 Temperature for Thermal Analyses

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 thermal stress-free state for the piping systems is defined as a temperature of 21°C (70°F) unless a basis is provided to use a higher temperature in accordance with ASME BPVC, Section III.

Applicable equipment nozzle movements are considered for their effect with respect to each operating mode. Radial and vertical movements of the RPV nozzles are considered in the thermal analysis.

For Class 1 piping systems the moduli of elasticity are listed in ASME BPVC, Section II, “Materials,” (Reference 3A-132), Part D, Subpart 2, Table TM.

Flexibility calculations of the forces and moments in a piping system due to thermal expansion and end motions are based on the hot modulus of elasticity E_h. However, the expansion stresses computed from the forces and moments are multiplied by the ratio of the cold modulus of elasticity E_c and the hot modulus, E_c/E_h in accordance with ASME BPVC, Section III, Division 1, Subparagraph NB-3672.5.

For Class 2 and 3 piping systems, the moduli of elasticity to be used are listed in the applicable tables of ASME BPVC, Section II (Reference 3A-132).

For non-nuclear (ASME B31.1) piping systems, the moduli of elasticity to be used are listed in ASME B31.1, Appendix C (Reference 3A-106).

For non-nuclear (ASME B31.3) piping systems, the moduli of elasticity to be used are listed in ASME B31.3, Appendix C (Reference 3A-107).

Calculation of expansion stress is based on the modulus of elasticity at room temperature.

3A.6.9.3.18 Intersystem Loss-of-Coolant Accident

The design includes consideration of high energy and moderate energy fluid system piping located inside and outside of containment. Design bases and measures used to protect these SSCs, referred to in the following sections as essential SSCs, are discussed in Section 3A.4.4.

To the extent practical, low-pressure systems (e.g., systems and subsystems connected to the RCS which extend outside the primary containment boundary and including valve stem seals, pump seals, HX tubes) are designed to withstand full RCS pressure. For those interfacing systems or subsystems which do not meet the full RCS Ultimate Rupture Strength requirement, the Plant Designer determines by evaluation that the degree and quality of isolation or reduced severity of the potential pressure challenges are low enough to preclude an intersystem LOCA.

3A.6.9.3.19 Effects of Environment on Fatigue Design

Refer to Section 3A.6.3.1 which describes the design bases which includes the fatigue analysis.

Class 1 piping is evaluated for the effects of fatigue as a result of pressure and thermal transients and other cyclic events including earthquakes.

Without considering exemption for the seismic fatigue analysis, the seismic fatigue usage factor is determined from Equations (11 and 14) in ASME BPVC, Section III, Division 1, Subparagraphs NB-3653.2 and NB-3653.6, respectively. This fatigue factor is added to the fatigue usage factor associated with Levels A and B load ranges to obtain the cumulative fatigue usage factor with the minimum number of cycles for the maximum peak response levels taken as 25 Cycles for floor-supported piping and 20 Cycles for ground-supported systems.

Results of the usage factor is calculated in accordance with ASME BPVC, Section III, Division 1, Subparagraph NB-3653.5 combined with the environmental effects per RG 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”.

3A.6.9.4 Piping Support Design

Pipe support types include anchor, spring hanger, snubber, strut, and supporting steel frame.

Anchor type supports provide six-way restraints to a pipe. Spring type supports are used to carry the dead weight of a pipe. Snubber type supports restrain a pipe in a seismic event but allow thermal displacement during normal operation. Strut type supports restrain pipe movements for all conditions. Supporting frames are usually used to support a standard support item and/or to restrain a pipe in a single or in multiple directions.

Pipe Whip Restraints are not in the jurisdictional boundary of ASME BPVC, Section III, Division 1, Subsection NF (Section 3A-123), ASME B31.1 (Reference 3A-106), or ASME B31.3 (Reference 3A-107) codes, and thus are not considered as pipe supports and they are not within the scope of this document.

Pipe supports are designed and qualified to satisfy stiffness values used in the piping analysis. For struts and snubbers, the stiffness to consider is the combined stiffness of strut, snubber, pipe clamp, and piping support steel.

Steam line supports are designed for water-filled line loads under static loading conditions that may be encountered in plant operations. In addition, provisions are made for conveniently supporting the deadweight loads imposed during hydrostatic test of the Main Steam piping.

The stiffness of the building steel/structure (i.e., beyond the NF jurisdictional boundary) is not considered in pipe support overall stiffness. Response spectra input to the piping system includes flexibility of the building structure. When attachment to a building structure is not possible, any intermediate structures are included in the analysis of the pipe support.

For multi-supported systems subjected to Independent Support Motion, the Independent Support Motion method of analysis can also be performed using the time history method.

3A.6.9.4.1 Pipe Support Applicable Codes

The pipe supports attached to the ASME Class 1 piping meet the appropriate requirements of ASME BPVC, Section III, Division 1, Subsection NF (Reference 3A-97).

The pipe supports attached to the ASME Class 2 and 3 piping meet the appropriate requirements of ASME BPVC, Section III, Division 1, Subsection NF (Reference 3A-97).

The pipe supports attached to the non-nuclear (ASME B31.1) piping meet the requirements of Sections 120 and 121 of ASME B31.1 “Power Piping” (Reference 3A-106).

The pipe supports attached to the non-nuclear (ASME B31.3) piping meet the requirements of Section 321 of ASME B31.3, “Process Piping” (Reference 3A-107).

3A.6.9.4.2 Pipe Support Jurisdictional Boundaries

Support design jurisdictional boundaries at interfaces with piping, structure, or intervening elements are defined in ASME BPVC, Section III, Division 1, Subsection NF, NF1130 (Reference 3A-97). If piping supports transmit loads to surface-mounted baseplates as discussed in ASME BPVC, Section III, Division 1, Subparagraph NF-1132(d) (Reference 3A-123), the baseplates are within the building structure jurisdiction. Refer to Section 3A.6.3.18.

3A.6.9.4.3 Pipe Support Loads and Load Combinations

The design loading is established considering plant and system operating conditions anticipated or postulated to occur during the intended service life of the component, system, or structure. The criteria are described in Section 3A.6.3.18.

Component and system piping loads and load combinations are provided in Table 3A-15, Table 3A-16, Table 3A-17, Table 3A-18, Table 3A-19 and Table 3A-20.

3A.6.9.4.4 Pipe Support Baseplate and Anchor Bolt Design

All aspects of the anchor bolt design, including baseplate flexibility and factors of safety are used in the development of anchor bolt loads, as addressed in USNRC Bulletin 79-02, Rev.2, “Pipe Support Base Plate Design Using Concrete Expansion Anchor Bolts,” (Reference 3A-133), and per ASME BPVC, Section III, Division 1, Subparagraph NF-1132(d) (Reference 3A-123).

3A.6.9.4.5 Use of Energy Absorbers and Limit Stops

ASME Section III component supports are designed, manufactured, installed, and tested in accordance with 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. If energy absorbers and limit stops are required for use in the piping system, the design methodology and acceptance criteria are defined and approved. The evaluation consists of iterative response spectra analyses of the piping and support system.

3A.6.9.4.6 Use of Snubbers

Snubber type supports are used when restraint is needed for dynamic type loading such as seismic or relief valve discharge at a location in the pipe that has large thermal movement. Snubbers are only active during dynamic events. Snubbers allow free thermal movement during normal operation and therefore are not loaded during normal operation. Due to maintenance concerns, the use of snubbers is minimized or avoided.

Snubbers are modelled with an equivalent stiffness based on dynamic tests or on data provided from the vendor. The permissible loads that the piping system can place on snubbers is defined in ASME BPVC, Section III, Division 1, Subsection NF (Reference 3A-97).

The load combinations and acceptance criteria applied to the analysis of snubber type supports.

Note: Snubbers are only active during dynamic loading. The design is not detailed enough to determine if the use of snubbers is required.

3A.6.9.4.7 Pipe Support Stiffness

All rigid supports are sized and qualified for a deflection limit of 3.2 mm (1/8 in.) for all conditions in the restraint directions. Where the support deflection cannot meet the minimum criteria, the actual support stiffness is developed and either justified as sufficiently rigid or included in the piping analysis. In the event the actual stiffness is included, all supports within the stress case boundary are evaluated with actual stiffness values.

3A.6.9.4.8 Seismic Self-Weight Excitation

For the component type supports (e.g., snubbers, struts, spring hangers), support component weights may be neglected if it is demonstrated that the support is dynamically rigid or that one half of the support mass is less than 10% of the mass of the straight pipe segment of the span at the support location, to preclude amplification. Otherwise, piping supports are evaluated to include the effect of self-weight excitation on the support structure, and anchorage in detail along with piping analysed loads where this effect is significant.

Section 3A.3.1.3 discusses the seismic analysis methodology used to analyse all BWRX-300 Seismic Category 1A or 1B components, piping, and their supports to function as needed during and after an earthquake.

3A.6.9.4.9 Design of Supplementary Steel

The design of structural steel for use as frame type pipe supports are linear supports as defined in ASME Code, Section III, Subsection NF (Reference 3A-97).

3A.6.9.4.10 Consideration of Friction Forces

The friction force on the pipe during axial movement is considered in the design of the support.

3A.6.9.4.11 Pipe Support Gaps and Clearances

Small gaps are provided for frame type supports allowing for radial thermal expansion of the pipe without imposing any thermal binding and to restrain the pipe.

A gap of 1.6 mm (1/16 in) is provided in restraint directions except along the gravity direction. The total gap between the pipe and the frame support is 3.2 mm (1/8 in) along the non-gravity direction. For large pipes with higher temperatures, gaps are evaluated to assure that no thermal binding occurs. The cold gap along unrestraint directions is 50 mm (2 in) more than the maximum displacement of the pipe.

3A.6.9.4.12 Instrumentation Line Support Criteria

The instrumentation line supports attached to the ASME Class 1 piping meet the appropriate requirements of ASME BPVC, Section III, Division 1, Subsection NF (Reference 3A-97).

Section 3A.6.1.4 discusses the instrumentation line support criteria for faulted conditions. Design loads and load combinations for fixed mechanical equipment are provided in Table 3A-11 which also applies to BWRX-300 Seismic Category 1A or 1B instrumentation and electrical equipment.

3A.6.9.4.13 Pipe Deflection Limits

Rigid supports are sized and qualified for a deflection limit of 3.2 mm (1/8 in) for all conditions in the restraint directions. Where the support deflection cannot meet the minimum criteria, the actual support stiffness is developed and either justified as sufficiently rigid or included in the piping analysis. In the event the actual stiffness is included, all supports within the stress case boundary are evaluated with actual stiffness values.

3A.6.9.4.14 Clamp-Induced Local Pipe Stress Evaluation

The piping system and pipe supports are designed and located to minimize the necessity for welded attachments to the pipe. Where welded attachments exist, local stress evaluations are performed. The code requirements are met with the local stress added to the regular piping stress.

The local stresses are evaluated per Nonmandatory Appendix Y of the ASME Code (Reference 3A-122).

Use of stiff pipe clamps may result in local pipe stresses. Stiff pipe clamps described in Issue 89 of NUREG-0933, “Resolution of Generic Safety Issues (29),” (Reference 3A-134) are not to be used.

3A.6.10 Threaded Fasteners – ASME Code Class 1, 2 and 3

RCPB fasteners are designed, fabricated, erected, and tested to the highest quality standards practical, to have an extremely low probability of abnormal leakage/rapidly propagating failure/gross rupture. Designs include sufficient margin to assure that when stressed under operating, maintenance, testing, and postulated accident conditions; (A) the boundary behaves in a nonbrittle manner, and (B), the probability of rapidly propagating fracture is minimized and reflect consideration of service temperatures, and other conditions of the boundary material under operating, maintenance, testing, and postulated accident conditions and the uncertainties in determining; (1) material properties, (2) the effects of irradiation on material properties, (3) residual, steady state and transient stresses, and (4) size of flaws.

The BWRX-300 design conforms to the relevant requirements:

- Structures, systems, and components important to safety are designed, fabricated, erected, constructed, tested, and inspected to quality standards commensurate with the importance of the Safety Category function to be performed.
- The RCPB is designed, erected, and tested in a manner that provides assurance of an extremely low probability of abnormal leakage, rapidly propagating failure, or gross rupture.
- Cleaning of material and equipment is controlled to prevent damage or deterioration.

- The selection of materials, design, testing, fabrication, installation and inspection of threaded fasteners and mechanical joints are acceptable if they meet the criteria of the ASME Code, Section III, Class 1, 2, and 3 components.

3A.6.10.1 Design Aspects

The design of threaded fasteners (e.g., bolts, studs) complies with ASME BPVC, Section III, Articles NB-3000, NCD-3000, NE-3000, NF-3000, and NG-3000. Because ASME BPVC Subsection NE-3000 does not explicitly state the loads and load combinations that should be considered in the design of metal containment components, the load combinations and associated load factors comply with RG 1.57, “Design Limits and Loading Combinations for Metal Primary Reactor Containment System Components,” (Reference 3A-135). Fabrication of threaded fasteners complies with ASME BPVC, Section III, Articles NB-4000, NCD-4000, NE-4000, NF-4000, and NG-4000. Inspection of threaded fasteners complies with ASME BPVC, Section III, Subarticles NB-2500, NCD-2500, NE-5000, NF-2500 and NG-2500.

3A.6.10.1.1 Materials Selection

Material used for threaded fasteners (e.g., bolts, studs) complies with the requirements of ASME BPVC, Section III, Subsection NCA; ASME BPVC Section III, Division 1, Paragraph NB-2128 (Class I); ASME BPVC Section III, Division 1, Paragraph NCD-2128 (Class 2 and 3); ASME BPVC Section III, Division 1, Paragraph NF-2128 (supports); or ASME BPVC Section III, Division 1, Paragraph NG-2121(core support). Material for nuts conforms to SA-194 or to the requirements of one of the specifications for nuts or bolting listed in ASME BPVC Section II, Part D, Subpart 1, Table 4 (Class I) or Table 3 (Class 2 and 3).

Note: Paragraph NB-2122/NCD-2122 states that special requirements in Article NB/NCD-2000 applies in lieu of the requirements of the material specification (NCA-4256) when there is conflict between the two.

Fracture toughness testing is performed in accordance with ASME BPVC, Section III, Division 1, Paragraph NB-2021, Subarticle NB-2300, Paragraph NCD-2021, Subarticle NCD-2300, Subarticle NF-2300, Paragraph NG-2021, or Subarticle NG-2300, (bolting, including studs with a nominal thickness of greater than 1 inch). ASME BPVC, Section II, Part D, Section 5.2, and BPVC Section II, Part D, Subpart 1, provides the properties for various classes of bolting materials as follows:

Refer to RG 1.65, Rev. 1, “Materials and Inspections for Reactor Vessel Closure Studs,” (Reference 3A-136), for guidance on reactor vessel closure stud bolting.

Criteria for selection and testing of bolting materials is listed in Table 3A-25.

ASME Code Class 1 Applications

PSR Chapter 4 (Reference 3A-3) describes the use of threaded fasteners for tie rods lower end plugs, upper tie plate to accept channel fastener bolt, and for the FMCRD middle flange attachments to CRD housing.

PSR Chapter 5 (Reference 3A-4) discusses the specifications for pressure-retaining ferritic materials, nonferrous metals, and austenitic stainless steels, including bolting and weld materials, which are used for each component (vessels, piping, pumps, and valves) of the RCPB. The adequacy and suitability of the ferritic materials, stainless steels, and nonferrous metals specified for the above applications are discussed in PSR Chapter 5 (Reference 3A-4). PSR Chapter 5 (Reference 3A-4) lists the materials and ASME standards for the RCPB which include the Main Steam pipe, body bolting and bolting nuts.

ASME BPVC, Section II, Part D, provides the properties for the following Class 1 bolting materials:

- Class 1 Bolting: Table 4

- Class 1 Bolting for Core Support Structures: Table 2A, 2B

ASME Code Class 2 and 3 Applications

ASME BPVC, Section II, Part D, provides the properties for the following Class 2 and 3 bolting materials:

- Class 2 Bolting: Table 3
- Class 3 Bolting: Table 3

3A.6.10.1.2 Mechanical Testing, Special Process and Controls

The appropriate use and selection of lubricants or sealants for threaded fasteners conform to the recommended practices provided in NUREG-1339, “Resolution of Generic Safety Issue 29: Bolting Degradation or Failure in Nuclear Power Plants,” (Reference 3A-137). Note that molybdenum sulphide lubricant is known to promote corrosion in low alloy steel and is not used.

Measures are to be established to control the cleaning of material and equipment to prevent damage or deterioration. RG 1.28, “Quality Assurance Program Criteria,” (Reference 3A-138) provides quality assurance criteria for cleaning fluid systems and associated components. Application of the cleaning criteria in threaded fasteners provides assurance that contaminants to which they could be exposed will not damage or deteriorate the materials, alter their properties, accelerate effects associated with aging, or increase the susceptibility to failure mechanisms such as stress corrosion cracking. Application of these criteria reduces the likelihood that degradation of threaded fasteners could lead to loss of pressure boundary integrity.

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.

3A.6.10.1.3 Fracture Toughness Requirements for Ferritic Materials

Pressure-retaining ferritic material, and material welded thereto are impact tested in accordance with the requirements of ASME BPVC, Section III, Division 1, Subarticle NCD-2300 and Subarticle NCD-2400 to ensure adequate fracture toughness properties. Fracture toughness and the pressure-temperature limits of the RCPB and reactor vessel are discussed in PSR Chapter 5 (Reference 3A-4).

The fracture toughness of ferritic bolts, studs, and nuts (i.e., made from either low-alloy steel or carbon steel materials) is acceptable if the ASME Code, Section III criteria are appropriately specified by the applicant for ASME Code Class 1, 2, and 3 systems.

3A.6.10.1.4 Fabrication Inspection

Fabrication of threaded fasteners complies with:

- ASME BPVC, Section III, Division 1, Article NB-4000 for Class 1 components
- ASME BPVC, Section III, Division 1, Article NCD-4000 for Class 2 and 3 components
- ASME BPVC, Section III, Division 1, Article NE-4000
- ASME BPVC, Section III, Division 1, Article NF-4000
- ASME BPVC, Section III, Division 1, Article NG-4000

Inspection of threaded fasteners complies with:

- ASME BPVC, Section III, Division 1, Subarticle NB-2500 for Class 1 components

- ASME BPVC, Section III, Division 1, Subarticle NCD-2500 for Class 2 and 3 components
- ASME BPVC, Section III, Division 1, Article NE-4000
- ASME BPVC, Section III, Division 1, Subarticle NF-2500
- ASME BPVC, Section III, Division 1, Subarticle NG-2500

Note: Applicants or licensees applying the provisions of ASME BPVC, Section III, Division 1, Paragraph NB–2582, Paragraph NC–2582, Paragraph ND–2582, Paragraph NE–2582, Paragraph NF–2582, Paragraph NG–2582 in the 2017 Edition of Section III through the latest edition and addenda incorporated by reference in paragraph (a) (1) (i) of this section, must apply paragraphs (b) (1) (x) (A) and (B) of this section.

The visual examinations are required to be performed in accordance with procedures qualified to:

- ASME BPVC, Section III, Division 1, Subarticle NB-5100 for Class 1 components
- ASME BPVC, Section III, Division 1, Subarticle NCD-5100 for Class 2 and 3 components
- ASME BPVC, Section III, Division 1, Subarticle NE-5100
- ASME BPVC, Section III, Division 1, Subarticle NF-5100
- ASME BPVC, Section III, Division 1, Subarticle NG-5100

The visual examinations are performed by personnel qualified in accordance with:

- ASME BPVC, Section III, Division 1, Subarticle NB-5500 for Class 1 components
- ASME BPVC, Section III, Division 1, Subarticle NCD-5500 for Class 2 and 3 components
- ASME BPVC, Section III, Division 1, Subarticle NE-5500
- ASME BPVC, Section III, Division 1, Subarticle NF-5500
- ASME BPVC, Section III, Division 1, Subarticle NG-5500

Bolts, studs, and nuts are visually examined for discontinuities including cracks, bursts, seams, folds, thread lap, voids, and tool marks.

3A.6.10.1.5 Quality Records

For threaded fasteners, documentation related to fracture toughness (as applicable) and CMTRs 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 in accordance with ASME BPVC, Section III, Division 1, Paragraph NCA-2021, Paragraph NCA-1224.

3A.6.10.2 Pre-Service and In-Service Inspection Requirements

PSR Chapter 5 (Reference 3A-4) and PSR Chapter 6 (Reference 3A-5) describes the ISI program of ASME Class 1, 2 and 3 components.

3A.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 SC DCIS segments that have appropriate levels of hardware and software quality corresponding to the system functions they control and their allocation to the Defence 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 PSR Chapter 7 Section 7.1.1 (Reference 3A-6). The classification of I&C systems is described in PSR Chapter 7 Section 7.1.2 (Reference 3A-6) and is based on the general classification criteria described in Section 3.2.1. The I&C system of systems, and the individual I&C systems are described in PSR Chapter 7 Section 7.3 (Reference 3A-6).

3A.7.1 Performance

The system design bases, and associated safety functions, are described for the DL3 systems, DL4a systems, DL2 systems, and for the non-classified systems in PSR Chapter 7 Section 7.3.1, Section 7.3.2, Section 7.3.3 and Section 7.3.4 respectively (Reference 3A-6).

3A.7.2 Design for Reliability

The system reliability requirements and associated design features are described for the DL3 systems, DL4a systems, DL2 systems, and for the non-classified systems in PSR Chapter 7 Section 7.3.1, Section 7.3.2, Section 7.3.3 and Section 7.3.4 respectively (Reference 3A-6).

3A.7.3 Independence

The system independence requirements and associated design features are described for the DL3 systems, DL4a systems, DL2 systems and non-classified systems in PSR Chapter 7 (under the Robustness heading) Section 7.3.1, Section 7.3.2, Section 7.3.3 and Section 7.3.4 respectively (Reference 3A-6).

3A.7.4 Qualification

The system qualification requirements are described for the DL3 systems, DL4a systems, DL2 systems and for the non-classified systems in PSR Chapter 7 (under the Equipment Qualification heading) Section 7.3.1, Section 7.3.2, Section 7.3.3 and Section 7.3.4 respectively (Reference 3A-6).

3A.7.5 Verification and Validation

The system verification and validation requirements for I&C systems are described in PSR Chapter 7 Section 7.4.3 (Reference 3A-6).

3A.7.6 Failure Modes

The application of the single failure criterion to DL3 systems is described in PSR Chapter 7 Section 7.3.1 (Reference 3A-6). The effects of failures and associated design features to minimize or eliminate adverse effects of anticipated failures are described for the DL4a systems, DL2 systems and for the non-classified systems in PSR Chapter 7 Section 7.3.2, Section 7.3.3 and Section 7.3.4 respectively (Reference 3A-6).

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, DL4a systems, DL2 systems and for the non-classified systems in PSR Chapter 7 Section 7.3.1, Section 7.3.2, Section 7.3.3 and Section 7.3.4 respectively (Reference 3A-6).

3A.7.7 Control of Access to Equipment

The system security requirements (including control of access) are described for the DL3 systems, DL4a systems, DL2 systems and for non-classified systems in PSR Chapter 7 Section 7.3.1, Section 7.3.2, Section 7.3.3 and Section 7.3.4 respectively (Reference 3A-6).

3A.7.8 Quality

The industry standards and USNRC Regulatory Requirements used for the I&C systems are described in PSR Chapter 7 Section 7.1.3 (Reference 3A-6), along with the system quality requirements.

3A.7.9 Testing and Testability

The system testing requirements (including design features to support testability) are described for the DL3 systems, DL4a systems, DL2 systems and for the non-classified systems in PSR Chapter 7 Section 7.3.1, Section 7.3.2, Section 7.3.3 and Section 7.3.4 respectively (Reference 3A-6).

3A.7.10 Maintainability

The system maintainability requirements and associated design features are described for the DL3 systems, DL4a systems, DL2 systems and for the non-classified systems in PSR Chapter 7 Section 7.3.1, Section 7.3.2, Section 7.3.3 and Section 7.3.4 respectively (Reference 3A-6).

3A.7.11 Identification of Items Important to Safety

The I&C system classification information is described in PSR Chapter 7 Section 7.1.2 (Reference 3A-6).

3A.8 General Design Aspects for Electrical Systems and Components

The electrical power system design is a 50 Hz Alternating Current (AC) power system, with 6.9 kV for the Medium Voltage (MV) level and 690 VAC (Volts Alternating Current), and 400/230 VAC for the Low Voltage (LV) level.

The BWRX-300 design minimises the reliance on electrical power to support safety category functions. The passive design of the plant us not dependent upon AC power sources including diesel generators, to mitigate a DBA. SC1 power is supplied from battery backed DC power, which has a coping period of 72 hours for all DBAs.

Refer to PSR Chapter 8 (Reference 3A-7) – Electrical Power for a detailed discussion on the electrical power systems for the BWRX-300.

The BWRX-300 has two connections to the switchyard. The Preferred Power System provides the interconnecting Electrical Distribution System (EDS) elements between the plant main generator, on-site power system, and the offsite power source. The preferred power sources for the plant being the two switchyard connections. Given the passive safety design of the BWRX-300 plant, only a Preferred Power System connection between the plant and switchyard is required. However, two sources of offsite power are provided to improve reliability and operational flexibility.

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 Generator Step Up Transformer, which is connected to the plant generator and the Unit Auxiliary Transformer (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 Reserve Auxiliary Transformer (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 (SDGs). Each SC3 LV bus is connected to a SDG 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.

3A.8.1 Redundancy

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 SDGs are provided to power the plant SC1, SC2 and SC3 loads.

Three divisions of SC1 Direct Current (DC) power are not only redundant to each other, but also have redundant UPSs in each division 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 DC load groups and one SC3 DC load group each with an Uninterruptible Power Supply (UPS) to provide power to the respective SC2 and SC3 loads.

There are three independent SC1 DC divisions with UPS to provide power to SC1 loads.

Redundancy for the BWRX-300 electrical power systems is discussed in more detail in PSR Chapter 8 (Reference 3A-7).

3A.8.2 Independence

As discussed above, in the event of total loss-of-offsite power sources two on-site SC3 SDGs 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 SDGs 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 DC load groups and one SC3 DC load group each with a UPS to provide power to the respective SC2 and SC3 loads.

There are three independent SC1 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 PSR Chapter 8 (Reference 3A-7) sections. Refer to PSR Chapter 8 (Reference 3A-7) for further discussion of this topic.

3A.8.3 Diversity

The EDS is designed along a D-in-D philosophy and along DLs. Section 2 provides a discussion on philosophy. The electrical systems are diverse from each based on DLs.

3A.8.4 Control and Monitoring

On-site and Off-site electrical power system controls and monitoring for the BWRX-300 will be accomplished by both Main and SCRs 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 PSR Chapter 8 (Reference 3A-7).

3A.8.5 Identification

Refer to PSR Chapter 8 (Reference 3A-7) for details on the electrical system safety classification and a description of the major electrical power system equipment.

3A.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 PSR Chapter 8 (Reference 3A-7).

3A.8.7 External Grid and Related Issues

PSR Chapter 8 (Reference 3A-7) (Electrical Power) Section 8.3 includes a description of an assessment of compliance of the BWRX-300 to the UK Grid Code.

The BWRX-300 design has been subject to a high-level assessment against the National Energy System Operator Grid code, which defines the technical requirements for connecting to and using the National Electricity Transmission System. This assessment was performed to provide confidence that there are no design ‘blockers’ to UK deployment.

During a potential future UK project, as part of PCSR production, the Requesting Party will carry out further studies and evaluate any gaps identified in the UK Grid Code Assessment, addressing any shortfalls against the UK’s Electrical Grid Code and ensuring that solutions reduce risks to ALARP. Confirmation of compliance, identified gaps, proposed mitigations, and further work will be detailed in future licencing actions.

Table 3A-1: Seismic Categories and Design Basis of BWRX-300 Structures

Structure	BWRX-300 Seismic Category/Evaluation	Design/Evaluation Basis	Design Basis Earthquake [1]	Limit State [2]
SCCV and Containment Steel Structures	BWRX-300 Seismic Category 1A	ASCE/SEI 43 and ASCE/SEI 4 for analysis ASME BPVC and DP-SC for design (Reference 3A-57)	DBE	LS-D
Containment Internal Structures	BWRX-300 Seismic Category 1A	ASCE/SEI 43 and ASCE/SEI 4 for analysis ANSI/AISC N690 and DP-SC for design (Reference 3A-57)	DBE	LS-D
RB Steel-Plate Composite and Steel Structures	BWRX-300 Seismic Category 1A			
RWB Structure	BWRX-300 Seismic Category RW	USNRC RG 1.143, ASCE/SEI 43 and ASCE/SEI 4 for analysis ACI 349 and NSI/AISC N690 for design	½ DBE [3]	LS-D
	To meet II/I Seismic Interaction requirements [4]	ASCE/SEI 43 and ASCE/SEI 4	DBE	LS-C
CB Structure	BWRX-300 Seismic Category 2	IBC or local building code		
	To meet II/I Seismic Interaction requirements [4]	ASCE/SEI 43 and ASCE/SEI 4	DBE	LS-C
TB Structure	BWRX-300 Seismic Category 2	IBC or local building code		
	To meet II/I Seismic Interaction requirements [4]	ASCE/SEI 43 and ASCE/SEI 4	DBE	LS-C
Service Building Structure	BWRX-300 Seismic Category 2	IBC or local building code		

Structure	BWRX-300 Seismic Category/Evaluation	Design/Evaluation Basis	Design Basis Earthquake (1)	Limit State (2)
	To meet II/I Seismic Interaction requirements (4)	ASCE/SEI 43 and ASCE/SEI 4	DBE	LS-C
Reactor Auxiliary Structures	BWRX-300 Seismic Category 2	IBC or local building code		
	To meet II/I Seismic Interaction requirements (4)	ASCE/SEI 43 and ASCE/SEI 4	DBE	LS-C
Equipment Structures	Non-Seismic (NS)	IBC or local building code		

Notes

(1). Design Basis Earthquake as defined in Section 3.3.1.

(2). Limit States per ASCE/SEI 43 where LS-D is essentially elastic response, and LS-C response with limited permanent deformations.

(3). The RWB is designed in accordance with the radioactive waste management requirements from USNRC RG 1.143.

(4). Proximity of this building to the Seismic Category 1A RB structure requires seismic interaction evaluation to DBE.

Table 3A-2: Seismic Damping Values for BWRX-300 Structures

Material	Critical Damping	
	OBE (%)	**DBE (%)**
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 3A-3: Seismic Damping Values for RPV and Internals

Component	Critical Damping	
	OBE (%)	DBE (%)
Reactor Vessel	2	4
Vessel Support Skirt	2	4
Shroud	2	4
Shroud Support Spring	2	-
Shroud Head & Separator	2	4
Fuel	4	6
CRD Guide Tubes	1	2
CRD Housing	1	2
CRD Restraint Springs	2 (vertical direction only)	-
Stabilizer and Bellows	2 (vertical direction only)	-
Welded Steel	-	4
Bolted Steel	-	7

Table 3A-4: Seismic Damping Values for Piping and Equipment

Structure or Component	OBE Damping	DBE Damping
Equipment and large-diameter piping system, pipe diameter greater than 12 in.	3	4
Small-diameter piping systems, diameter equal to or less than 12 in.	2	4
Welded steel or bolted steel with friction connections	3	4
Bolted steel structures with bearing connections	4	7

Table 3A-5: Design Extreme Wind-Generated Missile Spectrum

Building	**Reactor Building, Main Control Room, and Main Control Room to Secondary Control Room Egress(3)(4)**		**Radwaste Building(4)**
	Tornado Missile Spectrum (1)	Extreme Wind (Hurricane) Missile Spectrum (1)	Tornado Missile Spectrum
Missile Velocity:			
Horizontal Missile Velocity, V_h	41 m/s (135 ft/s) for Automobile and Pipe, 8 m/s (26 ft/s) for Solid Sphere	V_h per Table 2 of RG 1.221 by linear interpolation for Design Wind speed 89.4 m/s (200 mph) (5) 56 m/s (183 ft/s) for Automobile 43.5 m/s (142 ft/s) for Pipe 38 m/s (125 ft/s) for Sphere	V_h per Table 2 of RG 1.143 for Design Basis Tornado Wind speed 76 m/s (170 mph) (6) 30 m/s (100 ft/s) for Automobile and Pipe
Vertical Missile Velocity, V_y	27.5 m/s (90.5 ft/s) for Automobile and Pipe, 5.4 m/s (17.6 ft/s) for Solid Sphere	26 m/s (85.3 ft/s) for all missiles (Per note Table 2 of RG 1.221)	V_y per Table 2 of RG 1.143 for Design Basis Tornado Wind speed 76 m/s (170 mph) (6) 20 m/s (67 ft/s) for Automobile & Pipe
Missile Characteristics:			
High Kinetic Energy Missile	Automobile 1810 kg (4000 lb) 5m x 2m x 1.3m (16.4 ft x 6.6 ft x 4.3 ft) C_DA/m (7) = 0.0070 m^2/kg (0.0343 ft^2/lb)	Automobile 1810 kg (4000 lb) 5m x 2m x 1.3m (16.4 ft x 6.6 ft x 4.3 ft)	Automobile 1810 kg (4000 lb) Frontal Area = 1.86 m^2 (20 ft^2)
Rigid Missile to Test Penetration Resistance	Schedule 40 Pipe 130 kg (287 lb) 0.168 m diameter x 4.58 m long (6.625 in diameter x 15 ft long), C_DA/m (7) = 0.0043 m^2/kg (0.0212 ft^2/lb)	Schedule 40 Pipe 130 kg (287 lb) 0.168 m diameter x 4.58 m long (6.625 in diameter x 15 ft long)	Schedule 40 Pipe (8) 130 kg (287 lb) 0.168 m diameter x 4.58m long (6.625 in diameter x 15 ft long)

Building	Reactor Building, Main Control Room, and Main Control Room to Secondary Control Room Egress(3)(4)		Radwaste Building(4)
	Tornado Missile Spectrum (1)	Extreme Wind (Hurricane) Missile Spectrum (1)	Tornado Missile Spectrum
Small Rigid Missile to Test Opening Size	Solid Steel Sphere 0.0669 kg (0.147 lb) 2.54 cm (1 in) diameter C_DA/m (7) = 0.0034 m^2/kg (0.0166 ft^2/lb)	Solid Steel Sphere 0.0669 kg (0.147 lb) 2.54 cm (1 in) diameter	Not Applicable (9)
Applicable Height of Missile	Automobile: Up to 9.14 m (30 ft) above grade within 0.5 mile (0.8 kilometer) of the plant structures. Pipe and Sphere: Full height of building	Automobile: Up to 9.14 m (30 ft) above grade within 0.5 mile (0.8 kilometer) of the plant structures. Pipe and Sphere: Full height of building	Automobile: Up to 9.14 m (30 ft) above grade. Pipe: Full height of building
Applicable RG	RG 1.76(2)	RG 1.221	RG 1.143 and ANSI-2.3 (6)

Notes:

Design for missile impact uses the representative missile speeds.

[[Values are for Region I from RG 1.76.]]

Only for portions requiring storm hardening (at a minimum the MCR and MCR to SCR egress route)

Design basis wind speeds are nominal 3-second gust in (m/s) mph at 33 ft (10m) above ground over open terrain.

The design wind speed used for determination of missile velocities for the BWRX-300 is 89.4 m/s (200 mph).

Per Table 2 of RG 1.143, Design Basis tornado wind speed is based on Table 3 of ANSI/ANS-2.3 at a probability of 10^{-5}/yr consistent with RG 1.76. Horizontal and vertical velocity coefficients are based on Table 4 of ANSI/ANS-2.3.

C_DA/m missile flight parameter including drag coefficient (C_D) x projected frontal area of missile (A)/mass of missile (m).

Schedule 40 pipe missile per Table 4 of ANSI/ANS-2.3 and consistent with Table 2 of RG 1.76.

Per Table 2 of RG 1.143, the design-basis missiles for the Radwaste building do not include the small rigid missile.

Table 3A-6: Minimum Acceptable Barrier Thickness Requirements for Local Damage Protection Against Tornado Generated Missiles

Concrete Strength MPa (Psi)	Wall Thickness mm (in)	Roof Thickness mm (in)
20.7 (3000)	462 (18.2)	335 (13.2)
27.6 (4000)	429 (16.9)	312 (12.3)
34.5 (5000)	406 (16.0)	297 (11.7)

Table 3A-7: Dynamic Increase Factor

Material	Dynamic Increase Factor	
	Yield Strength	Ultimate Strength
Reinforced Concrete		
Concrete Axial and Flexural Compressive Strength	–	1.25
Concrete Shear Strength	–	1.10
Diagonal tension and direct shear for reinforcing steel (stirrups)	–	1.0
Diagonal tension and direct shear (punchout) and bond for reinforced concrete	–	1.0
Reinforcing Steel Grade 280 MPa (40 ksi)	1.2	1.05
Reinforcing Steel Grade 350 MPa (50 ksi)	1.15	1.05
Reinforcing Steel Grade 420 to 550 MPa (60 – 80 ksi)	1.10	1.05
Structural Steel		
Carbon Steel Plate	1.29	1.10
Stainless Steel Plate	1.18	1.0

Table 3A-8: Permissible Ductility Ratios-Steel and Reinforced Concrete

Barrier Material	Limit State	Permissible Ductility Ratio (μ_p)	Reference
Steel Structures Barriers	Tension(1)	≤ min (0.25 $^{\varepsilon_u}/_{\varepsilon_y}$, $^{0.1}/_{\varepsilon_y}$)	ANSI/AISC N690, “Specification for Safety-Related Steel Structures for Nuclear Facilities,” Table NB3.1
	Flexure(1)(2)–Steel plates	≤ 20	
	Flexure(1)(2)–Open sections (W, S, WT)	≤ 10	
	Flexure(1)(2)–Closed sections (pipe, box section)	≤ 20	
	Members where shear(1)(2) governs the design	≤ 5	
	Compression, applicable when $F_e \geq 4.5F_y$	≤ min $(\frac{0.225}{F_y/F_e}, \frac{\varepsilon_{st}}{\varepsilon_y}, 10)$	
Reinforced Concrete Structures	Flexure(3)–Steel beams, walls, and slabs	min $(\frac{0.05}{(\rho-\grave{\rho})}, 10)$ Rotational capacity r_θ in radians of any yield hinge is limited to (0.0065 d/c) but not to exceed 0.07 radians	ACI 349 Clause F.3.3 and F3.4
	Shear-controlled beams, walls, and slabs: shear is resisted by concrete alone	≤ 1.0	ACI 349 Clause F.3.7 as modified by RG 1.142
	Shear-controlled beams, walls, and slabs: shear is resisted by concrete and stirrups, headed bars, or bent bars	≤ 1.3	
	Shear-controlled beams, walls, and slabs: shear is resisted by stirrups alone	≤ 3.0	
	Compression and Flexural produced from impactive loads Case 1: When compression controls the design, as defined by a load-moment interaction diagram	≤ 1.0	ACI 349 Clause F.3.8

Barrier Material	Limit State	Permissible Ductility Ratio (μ_p)	Reference
	Compression and Flexural produced from impactive loads Case 2: for beams columns, walls, and slabs When the compression load does not exceed 0.1fc'Ag or one-third of that which would produce balanced strain conditions, whichever is smaller	$\min\left(\frac{0.05}{(\rho - \grave{\rho})}, 10\right)$ Rotational capacity r_θ in radians of any yield hinge is limited to $(0.0065\, d/c)$ but not to exceed 0.07 radians.	
	The permissible ductility ratio varies linearly for conditions between those specified in Case 1 and Case 2.		
	Axial compressive load	≤ 1.3	ACI 349 Clause F.3.9

Notes:

For net sections with ductile behaviour, the plastic resistance is based on yielding of the net section. For net sections with either brittle or limited ductile behaviour, the member's plastic resistance is based on yielding of the gross section provided that the net section's tensile rupture based available strength exceeds its gross section's yielding based available strength.

Accompanying compression force, if any, is less than the smaller of $0.1F_eA_g$ and $0.1F_yA_g$.

For flexure to control the design, the shear strength of a structural member is at least 20% greater than the flexural strength; otherwise, the ductility ratios given for shear-controlled limit state are to be used.

Where:

ε_u: strain corresponding to elongation at failure (rupture) using the value corresponding to an 8-in. (200 mm) long specimen.

ε_y: strain corresponding to nominal yield stress = ${}^{F_y}/_{E}$

F_y: specified minimum yield stress, (ksi)

E: Modulus of elasticity 29,000 ksi (200,000 MPa) for carbon steel

ε_{st}: strain corresponding to the onset of strain hardening

F_e: elastic buckling stress determined according to $F_e = \frac{\pi^2 E}{(L_c/r)^2}$, (ksi)

r: radius of gyration, (in)

L_c: Effective length of member for buckling about the minor axis, (in)

f_c': Concrete compressive strength (psi)

A_g: Gross cross-sectional Area (in^2)

Table 3A-9: Permissible Ductility Ratios – Steel-Plate Composite

Element Type	Controlling Behaviour (1) (2)	Superficial Damage	Limited, Moderate, Severe Damage Ductility μ_d	Limited Damage Support Rotation r_θ (deg)(4)	Moderate Damage Support Rotation r_θ (deg)	Severe Damage Support Rotation r_θ (deg)
SC Slabs and Walls (including DP-SC)	Out-of-Plane Flexure	Essentially Elastic Behavior (3)	10(4)	1	4	6
	Out-of-Plane Shear:					
	Ties or Diaphragms spaced at no more than ½ t_{sc}		1.3			
	Ties or Diaphragms spaced at more than ½ t_{sc}		1.0			
	Compression		1.0(5)			
SC Shear Walls, and Diaphragms (including DP-SC)	In-Plane Flexure (shear walls and diaphragms)	Essentially Elastic Behavior (3)	N/A	-	1.5	2
	In-Plane Shear (shear walls)		3.0			
	In-Plane Shear (diaphragms)		1.5			

Notes:

SC components (including DP-SC) are classified as flexure-controlled if their available strength for the limit state of flexural yielding is less than their available strength for the limit state of out-of-plane shear failure by at least 25%.

When impact/impulse loading results are in net tension, shear capacity of concrete is not considered.

Essentially elastic behaviour means elastic structural analysis using design strain acceptance criteria of 1% for steel plates and 0.35% for concrete in compression, which corresponds to superficial damage of elements. The permissible ductility ratio μ_d is 1.0.

The limit of ductility of 3 can be used in lieu of support rotations of 1 degree complying with Regulatory Position C11.1.6 of RG 1.243.

For additional information refer to Regulatory Position C11.1.5 of RG 1.243.

Table 3A-10: Applicable Pressure Boundary Codes and Standards

Code or Standard Number	Title/Description
ASME BPVC Section II	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	Non-Destructive 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 Inservice Inspection (ISI) 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
ASME B31T	Standard Toughness Requirements for Piping
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
ASME QME-1	Qualification of Active Mechanical Equipment Used in Nuclear Facilities

Table 3A-11: Load Combinations and Acceptance Criteria

Plant Event / Event Combination	Service Loading Combination	ASME Service Level
Design	$P_D + T_D + R_D$	Design
Normal Operation	N	A
Plant / System AOO (Upset)	N + AOO	B
Seismic (Fatigue Only)	N + DBE	B (Note 1)
Emergency Level Transient, ATWS	N + ATWS	C
Emergency Level Transient, ELT	N + ELT	C
Small Break LOCA	N + SBLOCA	C
Small Break LOCA + DBE	N + SBLOCA + DBE	D
Large Break LOCA + DBE	N + LBLOCA + DBE	D
Faulted Level Transient, FLT	N + FLT	D
Checking Level Earthquake	N + CLE	D
Test	$P_t + T_t + R_t$	Testing Limit

Notes:

1. Applies only to fatigue evaluation of ASME BPVC Class 1 components and core support structures.
2. Nomenclature:
   - AOO Loads for AOO event
   - CLE Checking Level Earthquake
   - D Dead Load
   - D_t Dead Load for Test Condition
   - DBE Design Basis Earthquake Loads (Equivalent to SSE)
   - ELT Emergency Level Transient
   - FLT Faulted Level Transient
   - LBLOCA Large Break Loss-of-Coolant Accident
   - N Normal Operation Loads
   - P_D Design Pressure
   - P_t Test Pressure
   - DBA Loads for DBA event
   - R_D Design Mechanical Loads
   - R_t Test Mechanical Loads
   - SBLOCA Small Break Loss-of-Coolant Accident
   - T_D Design Temperature
   - T_t Test Temperature

Table 3A-12: Design Condition Definition

Service Condition	American Society of Mechanical Engineers Service Level (Component)	Plant State or Estimated Frequency Occurrence of PIE (F, per Reactor Year) Used for Categorisation
Normal	A	Planned Operation
Upset	B	AOO
Emergency	C	DBA of low frequency
Faulted	D	DBA or DEC of very low frequency

Note:

1. Events with a probability of occurrence below approximately 1E-07 have a probability of occurrence that is low enough that the events do not need to be factored into the 60-year life of the plant. The probabilistic safety analyses may identify initiating events or event sequences with a probability of occurrence below 1E-07 where the consequences of those events present a significant risk to the plant. In these cases of significant risk, the event is included in the design. Additionally, some events may be required to be considered, per regulation, regardless of frequency. Designs to mitigate the response or consequences of those events or event sequences are to be addressed on an event-specific basis.

Table 3A-13: Cycles of Events

Event Number	Section Number	Description	Design Basis Number of Cycles	ASME Service Level
1	4.1.1	Bolt-up	72	A
2A	4.1.16	Hydrostatic Testing	6	A
2B	4.1.17	System Leakage Test	144	A
3	4.1.2	Startup	200	A
4	4.1.3	Low-Pressure Hot Standby	200	A
5	4.1.4	High-Pressure Hot Standby	200	A
6	4.1.5	Turbine Roll and Increase to Rated Power	200	A
7/8	4.1.6	Daily/Weekly Load Reduction and Recovery	20,820	A
9	4.1.7	Rod Sequence/Pattern Change	90	A
10	4.2.1	Loss of Feedwater Heaters – Partial	59	B
11	4.2.8	Inadvertent Isolation Condenser Initiation – One Train	10	B
12	4.2.9	Feedwater Flow Increase	19	B
13	4.2.3, 4.2.2	Turbine Generator Trip, Other Scrams with Bypass Flow	67	B
14	4.2.4	Turbine Valve Fail Open	25	B
15	4.2.5, 4.2.6, 4.2.7, 4.2.10	Scrams Without Bypass	67	B
16	4.1.18, 4.1.19, 4.1.20, 4.1.21, 4.1.22	Rated Power Operation	N/A	A
17	4.1.9	Reduction to 0% Power	200	A
18	4.1.10	Hot Shutdown	200	A
19	4.1.11	Stable Shutdown	200	A
20	4.1.12	Cold Shutdown	200	A
21/22	4.1.13	Vessel Flooding/Shutdown Cooling	72	A
23	4.1.14	Unbolt	72	A
24	4.1.15	Refuel	72	A
25	4.3.1	Loss of Feedwater Heaters – Total	1	C
26	4.3.2	Design Basis Accident Pressurization	1	C

Event Number	Section Number	Description	Design Basis Number of Cycles	ASME Service Level
27	4.3.3	Design Basis Accident Pressurization Without Feedwater	1	C
28	4.3.4	Feedwater Flow Increase – Both Pumps	1	C
29	4.3.5	Loss of Feedwater – Both Pumps	1	C
30	4.3.6	All Turbine Bypass Valves and Turbine Control Valves Fail Open	1	C
31	4.3.7	Inadvertent Boron Injection System Actuation	1	C
32	4.3.8	Inadvertent Isolation Condenser Initiation – All Trains	2	C
33	4.3.9	Small Pipe Rupture — Loss-of-Coolant Accident	1	C
34	4.4.1	Bounding Transient Without scram	1	D
35	4.4.2	Large Pipe Rupture – Loss-of-Coolant Accident	1	D
36	4.4.3	Ultimate Pressure Regulation	1	D

Table 3A-14: Preliminary Active Valve List

Valve Number * Denotes Multiple HCUs/FMCRDs	Quantity	Description	ASME Code Class
Nuclear Boiler System Valves			
B21-1-YRIV-0601-000	1	Reactor Pressure Vessel (RPV) Head Vent Inboard Isolation	1
B21-1-YRIV-0602-000	1	RPV Head Vent Outboard Isolation	1
B21-1-YRIV-0201-A00 B21-1-YRIV-0201-B00 B21-1-YRIV-0201-C00	3	ICS Supply Inboard Isolation	1
B21-1-YRIV-0202-A00 B21-1-YRIV-0202-B00 B21-1-YRIV-0202-C00	3	ICS Supply Outboard Isolation	1
B21-1-YRIV-0101-A00 B21-1-YRIV-0101-B00	2	Main Steam Inboard Isolation	1
B21-1-YRIV-0102-A00 B21-1-YRIV-0102-B00	2	Main Steam Outboard Isolation	1
B21-1-YRIV-0501-A00 B21-1-YRIV-0501-B00	2	Reactor Water Cleanup Inboard Isolation	1
B21-1-YRIV-0502-A00 B21-1-YRIV-0502-B00	2	Reactor Water Cleanup Outboard Isolation	1

Valve Number * Denotes Multiple HCUs/FMCRDs	Quantity	Description	ASME Code Class
B21-1-YRIV-0301-A00 B21-1-YRIV-0301-B00 B21-1-YRIV-0301-C00 B21-1-YRIV-0301-D00	4	Feed Water Inboard Isolation	1
B21-1-YRIV-0302-A00 B21-1-YRIV-0302-B00 B21-1-YRIV-0302-C00 B21-1-YRIV-0302-D00	4	Feed Water Outboard Isolation	1
B21-1-YRIV-0401-A00 B21-1-YRIV-0401-B00 B21-1-YRIV-0401-C00	3	ICS Return Inboard Isolation	1
B21-1-YRIV-0402-A00 B21-1-YRIV-0402-B00 B21-1-YRIV-0402-C00	3	ICS Return Outboard Isolation	1
B21-1-YCIV-0103-A00 B21-1-YCIV-0103-B00	2	Main Steam Containment Isolation	2
B21-3-YCKV-0904-000	1	RPV Leak Detection Excess Flow Check	2
B21-1-YCKV-0714-A00 B21-1-YCKV-0714-B00 B21-1-YCKV-0714-C00	3	Reference Leg Purge Water Check	2
B21-1-YCKV-0713-A00 B21-1-YCKV-0713-B00 B21-1-YCKV-0713-C00	3	Reference Leg Purge Water Check	2

Valve Number * Denotes Multiple HCUs/FMCRDs	Quantity	Description	ASME Code Class
B21-1-YCKV-0702-A00 B21-1-YCKV-0702-B00 B21-1-YCKV-0702-C00	3	Reference Leg Excess Flow Check	2
B21-1-YCKV-0802-A00 B21-1-YCKV-0802-B00 B21-1-YCKV-0802-C00	3	Variable Leg Excess Flow Check	2
B21-1-YCKV-0613-000	1	Head Vent RPV Level Excess Flow Check	2
B21-1-YCKV-0126-A01 B21-1-YCKV-0126-A02 B21-1-YCKV-0126-A03 B21-1-YCKV-0126-B01 B21-1-YCKV-0126-B02 B21-1-YCKV-0126-B03	6	A/B Main Steam Excess Flow Check	2
Process Radiation and Environmental Monitoring System Valves			
D11-1-YCKV-0158-000 D11-1-YCKV-0159-000 D11-1-YCKV-0160-000 D11-1-YCKV-0161-000 D11-1-YCKV-0162-000 D11-1-YCKV-0163-000 D11-1-YCKV-0164-000	7	Containment Monitoring Excess Flow Check Valves	2

Valve Number * Denotes Multiple HCUs/FMCRDs	Quantity	Description	ASME Code Class
Isolation Condenser System Valves			
E52-1-YSBV-0003-A00-1 E52-1-YSBV-0003-B00-1 E52-1-YSBV-0003-C00-1	3	IC Purge Line Inboard Isolation	1
E52-1-YSBV-0004-A00-2 E52-1-YSBV-0004-B00-2 E52-1-YSBV-0004-C00-2	3	IC Purge Line Outboard Isolation	1
E52-1-YCKV-5008-A00 E52-1-YCKV-5008-B00 E52-1-YCKV-5008-C00	3	IC Steam Supply Leak Detection Excess Flow Check	2
E52-1-YCKV-5009-A00 E52-1-YCKV-5009-B00 E52-1-YCKV-5009-C00	3	IC Steam Supply Leak Detection Excess Flow Check	2
E52-1-YCKV-5006-A00 E52-1-YCKV-5006-B00 E52-1-YCKV-5006-C00	3	IC Steam Supply Leak Detection Excess Flow Check	2
E52-1-YCKV-5007-A00 E52-1-YCKV-5007-B00 E52-1-YCKV-5007-C00	3	IC Steam Supply Leak Detection Excess Flow Check	2
E52-1-YCKV-5010-A00 E52-1-YCKV-5010-B00 E52-1-YCKV-5010-C00	3	IC Steam Supply Leak Detection Excess Flow Check	2

Valve Number * Denotes Multiple HCUs/FMCRDs	Quantity	Description	ASME Code Class
E52-1-YCKV-5011-A00 E52-1-YCKV-5011-B00 E52-1-YCKV-5011-C00	3	IC Steam Supply Leak Detection Excess Flow Check	2
E52-1-YCKV-5021-A00 E52-1-YCKV-5021-B00 E52-1-YCKV-5021-C00	3	IC Condensate Leak Detection Excess Flow Check	2
E52-1-YCKV-5020-A00 E52-1-YCKV-5020-B00 E52-1-YCKV-5020-C00	3	IC Condensate Leak Detection Excess Flow Check	2
E52-1-YCKV-5017-A00 E52-1-YCKV-5017-B00 E52-1-YCKV-5017-C00	3	IC Condensate Leak Detection Excess Flow Check	2
E52-1-YCKV-5016-A00 E52-1-YCKV-5016-B00 E52-1-YCKV-5016-C00	3	IC Condensate Leak Detection Excess Flow Check	2
E52-1-YCKV-5019-A00 E52-1-YCKV-5019-B00 E52-1-YCKV-5019-C00	3	IC Condensate Leak Detection Excess Flow Check	2
E52-1-YCKV-5018-A00 E52-1-YCKV-5018-B00 E52-1-YCKV-5018-C00	3	IC Condensate Leak Detection Excess Flow Check	2

Valve Number * Denotes Multiple HCUs/FMCRDs	Quantity	Description	ASME Code Class
E52-1-YPOV-0002-A00 E52-1-YPOV-0002-B00 E52-1-YPOV-0002-C00	3	Air Operated Condensate Return Isolation	1
E52-1-YHCV-0001-A00 E52-1-YHCV-0001-B00 E52-1-YHCV-0001-C00	3	Hydraulic Operated Condensate Return Isolation	1
E52-1-YPOV-0005-A00 E52-1-YPOV-0005-B00	2	Shutdown Cooling Inboard Isolation	1
E52-1-YPOV-0006-A00 E52-1-YPOV-0006-B00	2	Shutdown Cooling Outboard Isolation	1
Boron Injection System Valves			
G11-1-YCKV-0072-000	1	Boron Injection Outboard Discharge Check	1
G11-1-YABV-0071-000	1	Boron Injection Air Operated Outlet Isolation	1
Control Rod Drive System Valves			
G12-1-YCKV-**15-A00	29	HCU Charge Flow Check	2
G12-1-YACV-**26-A00	29	HCU SCRAM	2
G12-1-YCKV-**38-A00	29	HCU Purge Water Check	2
G12-1-YSBV-**48-A00	29	SCRAM Solenoid Valve	2
G12-1-FCRD-****-A01	57	FMCRD Ball Check Valve	1

Valve Number * Denotes Multiple HCUs/FMCRDs	Quantity	Description	ASME Code Class
IC Pools Cooling and Cleanup System Valves			
G20-1-YCKV-0042-A00 G20-1-YCKV-0042-B00 G20-1-YCKV-0042-C00	3	Isolation Condenser (IC) Inner Pool Check	1
G20-1-YCKV-0041-A00 G20-1-YCKV-0041-B00 G20-1-YCKV-0041-C00	3	IC Outer Pool Check	1
G20-1-YABV-0032-A00 G20-1-YABV-0032-B00 G20-1-YABV-0032-C00	3	IC Inner Pool Sparger Supply Isolation	2
G20-1-YABV-0508-000	1	IC Outer Pool Sparger Supply Isolation	2
G20-1-YABV-0006-000	1	IC Pool Suction Surge Tank Isolation	2
Shutdown Cooling System Valves			
		No Valves identified	
Reactor Water Cleanup System Valves			
G31-1-YABV-0001-000	1	Reactor Water Cleanup Containment Isolation	2
Fuel Pool Cooling and Cleanup System Valves			
		No Valves identified	
Liquid Waste Management System Valves			
		No Active Valves identified	

Valve Number * Denotes Multiple HCUs/FMCRDs	Quantity	Description	ASME Code Class
Offgas System Valves			
		No valves identified – Steam Supply via Turbine Auxiliary Steam Supply	
Condensate and Feedwater System Valves			
N21-1-YCIV-1018-A00 N21-1-YCIV-1018-B00	2	Feed Water Containment Isolation	2
N21-1-YABV-1017-A00 N21-1-YABV-1017-B00	2	Feed Water System Isolation	2
N21-1-YABV-1124-A00 N21-1-YABV-1124-B00	2	Shutdown Cooling System Isolation	2
Condensate Filters and Demineralizers System Valves			
		No valves identified	
Main Turbine Equipment Valves			
		No valves identified	
Moisture Separator Reheater System Valves			
		No valves identified	
Turbine Bypass System Valves			
		No valves identified	
Main Condenser and Auxiliaries Valves			

Valve Number * Denotes Multiple HCUs/FMCRDs	Quantity	Description	ASME Code Class
		No valves identified	
Circulating Water System Valves			
		No valves identified	
Chilled Water Equipment Valves			
P25-1-YCIV-1014-A00 P25-1-YCIV-1014-B00	2	Containment Cooling Supply Outboard Isolation	2
P25-1-YCIV-1019-A00 P25-1-YCIV-1019-B00	2	Containment Cooling Return Inboard Isolation	2
P25-1-YCKV-1015-A00 P25-1-YCKV-1015-B00	2	Containment Cooling Supply Inboard Check	2
P25-1-YCIV-1021-A00 P25-1-YCIV-1021-B00	2	Containment Cooling Return Outboard Isolation	2
P25-1-YSRV-1020-A00 P25-1-YSRV-1020-B00	2	Containment Cooling Outboard Relief	2
Plant Cooling Water System Valves			
		No valves identified	
Plant Pneumatics System Valves			
P52-1-YCIV-1017	1	Outboard Nitrogen Supply Containment Isolation	2
P52-1-YCKV-1018	1	Inboard Nitrogen Supply Containment Check	2

Valve Number * Denotes Multiple HCUs/FMCRDs	Quantity	Description	ASME Code Class
Reactor Water Chemistry System Valves			
		No valves identified – Hydrogen Supply to Feedwater	
Primary Containment System Valves			
T10-1-YABV-0002-A00 T10-1-YABV-0002-B00 T10-1-YABV-0002-C00	3	Passive Containment Cooling Inlet Maintenance Isolation	2
T10-1-YABV-0004-A00 T10-1-YABV-0004-B00 T10-1-YABV-0004-C00	3	Passive Containment Cooling Inboard Containment Isolation	2
T10-1-YABV-0007-A00 T10-1-YABV-0007-B00 T10-1-YABV-0007-C00	3	Passive Containment Cooling Outlet Containment Isolation	2
T10-1-YABV-0008-A00 T10-1-YABV-0008-B00 T10-1-YABV-0008-C00	3	Passive Containment Cooling Outlet Maintenance Isolation	2
Containment Inerting System Valves			
T31-1-YCKV-0065-A00	1	Equipment Pool Inlet Check	2
Containment Cooling System Valves			
		No valves identified	

Valve Number * Denotes Multiple HCUs/FMCRDs	Quantity	Description	ASME Code Class
Fire Protection System Valves			
		No valves identified	
Equipment and Floor Drain System Valves			
U50-1-YCIV-2011-000	1	Containment Sump Inlet Inboard Isolation	2
U50-1-YCIV-2010-000	1	Containment Sump Inlet Outboard Isolation	2
Water Gas and Chemical Pads Valves			
		No Active valves identified	

Table 3A-15: Load Combinations and Acceptance Criteria for Class 1 Piping

Condition	Load Combination for all Terms	Acceptance Criteria	ASME Code Reference
Design	Design Pressure + Weight + Sustained Mechanical Load (Note 1)	Eq 9 $\leq$ 1.5 S_m	NB-3652
Service Levels A and B (Note 2)	Peak Pressure + Weight + Sustained Mechanical Load + Thermal Expansion + AOO (Note 3) + SSE (Note 4)	Eq 10 $\leq$ 3.0 S_m or Eq 12 & 13 $\leq$ 3.0 S_m and Fatigue CUF < 1.0 and Range ΔT_1 ≤ Allowable	NB-3653.1 NB-3653.6 NB-3653.1 to NB-3653.6 NB-3653.7
Service Level B	Peak Pressure + Weight + AOO	Eq 9 $\leq$ 1.8 S_m, not greater than 1.5 S_y (Note 5) and Peak Pressure $\leq$ 1.1P_a	NB-3654.2(a) NB-3654.1
Service Level C	Peak Pressure + Weight + Design Basis Accident (DBA) Where DBA can be one of items as listed below: 1. Emergency Level Transient 2. Anticipated Transient Without Scram (ATWS) 3. Small Break Loss of Coolant Accident (SBLOCA)	Eq 9 $\leq$ 2.25 S_m, not greater than 1.8 S_y (Note 6) (Note 7) and Peak Pressure $\leq$ 1.5 P_a	NB-3655.2(a) NB-3655.1
Service Level D	Peak Pressure + Weight + [DBA or Design Extension Condition (DEC)] Where [DBA or DEC] can be one of items as listed below: 1. SBLOCA + SSE 2. Large Break LOCA + SSE 3. Faulted Level Transient 4. Checking Level Earthquake (CLE)	Eq 9 $\leq$ 3.0 S_m, not greater than 2.0 S_y (Note 8) (Note 9) and Peak Pressure $\leq$ 2.0 P_a (Note 9) and Stress caused by Seismic Anchor Movement due to SSE satisfies NB-3656(b)(4)	NB-656(a)(2) NB-656(a)(1) NB-656(b)(4)

Notes:

(1) Effect of sustained mechanical load is considered together with pressure and weight. In normal operation load, fluid flow (including Flow-Induced Vibration when applicable), thermal and fluid reaction forces, and the like, are considered together with pressure and weight.

(2) Load combination is required to set up individual load set. Acceptance criteria for stress range including Equations 10, 12, and 13 are checked for each pair of load sets. One of the load sets to be included is that corresponding to zero pressure, zero moment, and room temperature.

(3) An event category assigned to a PIE or event sequence with an expected frequency of 1/yr to 1.0 E-02/yr that represents a deviation from normal operation but does not evolve into a postulated accident because of DL2 functions.

(4) SSE is only used for piping fatigue analysis. For piping fatigue analysis, the full range of SSE maximum stress cycles are considered. Use 25 Cycles for floor-supported piping and 20 Cycles for ground-supported systems. SSE includes both Inertia Effect and Anchor Displacement Loads, which are combined using SRSS method.

(5) This criterion is for Service Level B loadings that do not include reversing dynamic loads (NB-3622.2) or have reversing dynamic loads combined with non-reversing dynamic load (NB-3622.4). For Level B loadings that include reversing dynamic loads that are not required to be combined with non-reversing dynamic loads, the above criterion for Levels A and B loadings applies.

(6) If the anchor motion due to reversing dynamic loads is excluded from NB-3654, it is assessed through the requirements of NB-3656(b)(4) using 70% of the allowable.

(7) As an alternative requirement for Level C loadings that include reversing dynamic loads that are not required to be combined with non-reversing dynamic loads, the requirements of NB-3656(b) is satisfied using the allowable stress in NB-3656(b)(2), 70% of the allowable stress in NB-3656(b)(3), and 70% of the allowable loads in NB-3656(b)(4). If the anchor motion due to reversing dynamic loads is excluded from Equation (9), it is assessed through the requirements of NB-3656(b)(4) using 70% of the allowable.

(8) If the anchor motion due to reversing dynamic loads is excluded from NB-3654, it is assessed through the requirements of NB-3656(b)(4).

(9) See alternative requirements in NB-3656(b).

(10) Nomenclature:

- AOO Anticipated Operational Occurrences
- DBA Design Basis Accident
- DEC Design Extension Condition Events
- P_a Calculated maximum allowable internal pressure for a straight pipe that is at least equal the Design Pressure
- S_A Allowable stress range for expansion stresses = f (1.25 S_c+ 0.25 S_h), where f = stress range eduction factor for cyclic conditions
- S_c Basis material allowable stress at minimum (cold) metal temperature
- S_h Basis material allowable stress at maximum (hot) metal temperature
- S_m For Eq 9, allowable design stress intensity value at a temperature consistent with the loading under consideration; for Eq 10, 12 and 13, average of the allowable stress intensity value for the highest and the lowest temperature of the metal during the transient.
- S_y Yield Strength
- SRSS Square-root-of-the-sum-of-the-squares
- TE Thermal Expansion Range in Normal and Upset Conditions
- ∆T1 Value of the range of the temperature difference between the temperature of the outside surface To and the temperature of the inside surface Ti of the piping product assuming moment generating equivalent linear temperature distribution.

Table 3A-16: Load Combinations and Acceptance Criteria for Class 2/3 Piping

Condition	Load Combination for all Terms	Acceptance Criteria	ASME Code Reference
Design	Design Pressure + Weight + Sustained Mechanical Load (Note 2)	Equation (8) $\leq$ 1.5 S_h	NCD-3652
Service Levels A and B	Thermal Expansion	Equation (10a) $\leq$ S_A (Note 1)	NCD-3653.2
Service Levels A and B	Single Non-repeated Anchor Movement	Equation (10b) $\leq$ 3.0 S_C	NCD-3653.2
Service levels A and B	Design Pressure + Weight + Sustained Mechanical Load + Thermal Expansion (Note 2)	Equation (11) $\leq$ (S_h + S_A) (Note 1)	NCD-3653.2
Service Level B	Peak Pressure + Weight + AOO (Note 3) Where AOO can be the item as listed below: 1. AOO Plant/System Upset	Equation (9) $\leq$ 1.8 S_h, but not greater than 1.5 S_y (Note 4)	NCD-3653.1
Service Level C	Peak Pressure + Weight + DBA Where DBA can be one of items as listed below: 1. Emergency Level Transient 2. ATWS 3. SBLOCA	Equation (9) $\leq$ 2.25 S_h, but not greater than 1.8 S_y (Note 5) (Note 6)	NCD-3654.2
Service Level C	Peak Pressure	Peak Pressure $\leq$ 1.5 P_a	NCD-3654.1
Service Level D	Peak Pressure + Weight + [DBA or DEC] Where [DBA or DEC] can be one of items as listed below: 1. SBLOCA + SSE 2. Large Break LOCA + SSE 3. Faulted Level Transient 4. CLE	Equation (9) $\leq$ 3.0 S_h, but not greater than 2.0 S_y (Note 7) (Note 8)	NCD-3655
Service Level D	Seismic Anchor Movement due to SSE	Equation in NCD-3655(b)(4)	NCD-3655
Service Level D	Peak Pressure	Peak Pressure $\leq$ 2.0 P_a	NCD-3655

Notes:

(1) Meet either Equation 10a or 11).

(2) Effect sustained mechanical load is considered together with pressure and weight. In normal operation load, fluid flow (including Flow-Induced Vibration when applicable), thermal and fluid reaction forces is considered together with pressure and weight.

(3) An event category assigned to a PIE or event sequence with an expected frequency of 1/year to 1.0 E-02/year that represents a deviation from normal operation but does not evolve into a postulated accident because of DL2 functions.

(4) If the anchor displacement due to dynamic loads is excluded from Equation (9), it is included in Equation (10a) or (11).

(5) Service Loadings, which Level C Service Limits are designated for, includes reversing dynamic loads that are not required to be combined with non-reversing dynamic loads.

(6) If the anchor motion due to reversing dynamic loads is excluded from Equation (9), it is assessed through the requirements of NCD-3655(b)(4) using 70% of the allowable.

(7) Service Loadings, which Level D Service Limits are designated for, includes reversing dynamic loads that are not required to be combined with non-reversing dynamic loads/

(8) If the anchor motion due to reversing dynamic loads is excluded from Equation (9), it is assessed through the requirement of NCD-3655(b)(4).

(9) Nomenclature:

- AOO Anticipated Operational Occurrences
- CLE Checking Level Earthquake
- DBA Design Basis Accident
- DEC Design Extension Condition Events
- P_a Calculated maximum allowable internal pressure for a straight pipe that is at least equal the Design pressure
- PD Design Pressure
- PP Peak Pressure Operating Pressure associated with Transient
- TE Thermal Expansion Range in Normal and Upset Conditions
- SBLOCA Small Block LOCA
- SSE Safe Shutdown Earthquake. Includes both inertial and displacement components which are combined using SRSS method.
- S_A Allowable stress range for expansion stresses = f (1.25 S_c+0.25 S_h), where f = stress range reduction factor for cyclic conditions.
- S_c Basic material allowable stress at minimum (cold) metal temperature
- S_h Basic material allowable stress at maximum (hot) metal temperature
- S_y Yield Strength

Table 3A-17: Load Combinations and Acceptance Criteria for ASME B31.1 Piping and Components

Description	Load Combination	Acceptance Criteria	Requirement
Sustained	Design Pressure + Weight + Other Mechanical Loads	Eq (15) ≤ 1.0 S_h	ASME B31.1-2022, Paragraph 104.8.1
Occasional	Design Pressure + Weight + Other Mechanical Loads + Seismic Seismic event includes but is not limited to: • SSE • CLE	Eq (16) ≤ k S_h (Note 2)	ASME B31.1-2022, Paragraph 104.8.2
Occasional	Design Pressure + Weight + Other Mechanical Loads + Occasional event other than Seismic Occasional event includes but is not limited to: • AOO Plant/System Upset (Note 3) • Emergency Level Transient • ATWS • SBLOCA • Large Break LOCA • Faulted Level Transient • Wind • Snow Load	Eq (16) ≤ k S_h (Note 2)	ASME B31.1-2022, Paragraph 104.8.2
Thermal Expansion and other Cyclic Load	Displacement Stress Range	Eq (17) ≤ f (0.25 S_h + 1.25 S_C (Note 1) (Note 5)	ASME B31.1-2022, Paragraphs 104.8.3 and 102.3.2(b)(1)
Noncyclic Displacement	Noncyclic Displacement Stress Range	Eq (17) ≤ 3.0 S_C (Note 4)	ASME B31.1-2022, Paragraph 102.3.2(b)(2)
Test	Test Pressure	Circumferential (hoop) stress ≤ 0.9 * Yield Stress	ASME B31.1-2022, Paragraph 102.3.3 (B)

Description	Load Combination	Acceptance Criteria	Requirement
Test	Test Pressure + Weight	Longitudinal stress ≤ 0.9 * Yield Stress	ASME B31.1-2022, Paragraph 102.3.3 (B)

Notes:

(1) Stated Eq (17) of ASME B31.1-2022 allowable stress is assuming a cycle stress range factor of 1 if total number of cycles is below 7,776. If total number cycles for displacement range exceeds 7,776 (ASME B31.1-2022, Paragraph 102.3.2 (b)(1c)) then a factor less than 1.0 is calculated per Eq (1C) in Section 102.3.2 and applied to the allowable stress.

(2) For conventional piping in accordance with ASME B31.1-2022 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.

(3) An event category assigned to a PIE or event sequence with an expected frequency of 1/year to 1.0 E-02/year that represents a deviation from normal operation but does not evolve into a postulated accident because of DL2 functions.

(4) Replace M_C with the moment range due to the noncyclic movement in Equation (17) to calculate the displacement stress range, S_E.

(5) When S_h is greater than S_L, the difference between them may be added to the term 0.25 S_h, and the allowable stress range is calculated by $S_A = f\ (1.25\ S_C + 1.25\ S_h - S_L)$.

(6) Nomenclature:

- Sh Basis material allowable stress at maximum (hot) metal temperature
- Sc Basic material allowable stress at minimum (cold) metal temperature
- Sy Yield Strength

Table 3A-18: Load Combinations and Acceptance Criteria for ASME B31.3 Piping and Components

Description	Load Combination	Acceptance Criteria	Requirement
Sustained	Design Pressure + Weight + Other Mechanical Loads	Eq (23a) ≤ 1.0 S_h (Note 5)	ASME B31.3-2022, Paragraph 320.2
Occasional	Design Pressure + Weight + Other Mechanical Loads + Seismic Seismic event includes but is not limited to: • SSE • CLE	Eq (23a) ≤ k S_h (Note 2) (Note 5)	ASME B31.3-2022, Paragraphs 302.3.6(a) and 320.2
Occasional	Design Pressure + Weight + Other Mechanical Loads + Occasional event other than Seismic Occasional event includes but is not limited to: • AOO Plant/System Upset (Note 4) • Emergency Level Transient • ATWS • SBLOCA • Large Break LOCA • Faulted Level Transient • Wind • Snow Load	Eq (23a) ≤ k S_h (Note 2)	ASME B31.3-2022, Paragraphs 302.3.6(a) and 320.2
Thermal and Settlement	Displacement Stress Range	Eq (17) ≤ f (0.25 S_h + 1.25 S_C) (Note 1) (Note 3) (Note 5) (Note 6) (Note 7)	ASME B31.3-2022, Paragraphs 319.4.4 and 302.3.5(d)
Test	Test Pressure	Circumferential (hoop) stress ≤ Yield Strength or 1.5 x Component Rating	ASME B31.3-2022, Paragraph 345.2.1(a)
Test	Test Pressure + Weight	Longitudinal stress ≤ Yield Strength or 1.5 x Component Rating	ASME B31.3-2022, Paragraph 345.2.1 (a)

Notes:

(1) Stated Eq (17) of ASME B31.3-2022 (Reference 3.12-7) allowable stress is assuming a cycle stress range factor of 1 if total number of cycles is below 7,776. If total number cycles for displacement range exceeds 7,776 (ASME B31.32022 (Reference 3.12-7), Paragraph 302.3.5(d)(1c)) then a factor less than 1.0 is calculated per Eq (1c) in Section 302.3.5 and applied to the allowable stress.

(2) For conventional piping in accordance with ASME B31.3-2022 (Reference 3.12-7) 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.

(3) The bending stress range used in Eq (17) for elbows, mitre bends, and branch connections are calculated in accordance with Eq (18) in Section 319.4.4(b).

(4) An event category assigned to a PIE or event sequence with an expected frequency of 1/year to 1.0 E-02/year that represents a deviation from normal operation but does not evolve into a postulated accident because of DL2 functions. Examples of AOO can be referred to 006N4873, Loading Criteria (Fixed Equipment Only) (2).

(5) The cold and hot allowable stress is calculated as defined in BPVC Section III – Rules for Construction of Nuclear Facility Components – Division 1 – Section NCD – Class 2 and Class 3 Components (19) per RG 1.143, Table 1 - Codes and Standards for the Design of SCC in Radwaste Facilities (20).

(6) The displacement stress range due to earth settlement can be greater than that permitted by the allowable stress if due consideration is given to avoidance of excessive localized strain and end reactions, per ASME B31.3-2022 (9), Paragraph 319.2.1(c).

(7) When Sh is greater than SL, the difference between them may be added to the term 0.25 Sh, and the allowable stress range is calculated by SA = f [1.25 (SC + Sh)– SL].

(8) Nomenclature:

- S_h Basic material allowable stress at maximum (hot) metal temperature
- S_c Basic material allowable stress at minimum (cold) metal temperature
- S_y Yield Strength

The fatigue usage limit is reduced for piping locations exempt from pipe break consideration.

Table 3A-19: Load Combinations and Acceptance Criteria for Snubber Type Supports

<table>
<tr><th>Condition</th><th>Load Combination for all Terms</th><th>Acceptance Criteria</th></tr>
<tr><td>Service Level B</td><td>AOO (Note 1)</td><td rowspan="3">Vendor Load Capacity Datasheet or Vendor Design Report Summary</td></tr>
<tr><td>Service Level C</td><td>DBA
Where DBA can be one of the items as listed below:
1. Emergency Level Transient
2. ATWS
3. SBLOCA</td></tr>
<tr><td>Service Level D</td><td>[DBA or DEC]
Where [DBA or DEC] can be one of the items as listed below:
1. SBLOCA + SSE
2. Large Break LOCA + SSE
3. Faulted Level Transient
4. CLE</td></tr>
</table>

Notes:

* Notes under Table 3A-15 and Table 3A-16 apply to Table 3A-19.

(1) An event category assigned to a PIE or event sequence with an expected frequency of 1/year to 1.0 E02/year that represents a deviation from normal operation but does not evolve into a postulated accident because of DL2 functions.

Table 3A-20: Load Combinations and Acceptance Criteria for Rigid Type Supports

Condition	Load Combination for all Terms	Acceptance Criteria
Service Level A	Weight + Thermal Expansion + Sustained Mechanical Load	Vendor Load Capacity Datasheet, Vendor Design Report Summary or Rules of NF3623
Service Level B	Weight + Thermal Expansion + AOO	
Service Level C	Weight + Thermal Expansion + DBA Where DBA can be one of the items as listed below: 1. Emergency Level Transient 2. ATWS 3. SBLOCA	
Service Level D	Weight + Thermal Expansion + [DBA or DEC] Where [DBA or DEC] can be one of the items as listed below: 1. SBLOCA + SSE 2. Large Break LOCA + SSE 3. Faulted level Transient 4. CLE	

Notes:

* Notes under Table 3A-15 and Table 3A-16 apply to Table 3A-20.

(1) Effect of sustained mechanical load is considered together with pressure and weight. In normal operation load, fluid flow (including Flow-Induced Vibration when applicable), thermal and fluid reaction forces, etc. is considered together with pressure and weight.

(2) An event category assigned to a PIE or event sequence with an expected frequency of 1/year to 1.0 E023/year that represents a deviation from normal operation but does not evolve into a postulated accident because of DL2 functions.

Table 3A-21: Load Combinations and Acceptance Criteria for ASME Class 1 Piping Connected to Reactor Pressure Vessel Nozzles

Condition	Load Combination for all Terms	Acceptance Criteria
Design	Design Pressure + Weight + Sustained Mechanical Load	In accordance with RPV and Nozzle Qualification Documentation/Specification
Service Level B	Peak Pressure + Weight + Sustained Mechanical Load + Thermal Expansion + AOO	
Fatigue	Peak Pressure + Weight + AOO +SSE	
Service Level C	Peak Pressure + Weight + DBA Where DBA can be one of the items as listed below: 1. Emergency Level Transient 2. ATWS 3. SBLOCA	
Service Level D	Peak Pressure + Weight + [DBA or DEC] Where [DBA or DEC] can be one of the items as listed below: 1. SBLOCA + SSE 2. Large Break LOCA + SSE 3. Faulted level Transient 4. CLE	

Note:

* Notes under Table 3A-15 and Table 3A-16 apply to Table 3A-21.

Table 3A-22: Load Combinations and Acceptance Criteria for ASME Class 2 and 3 Piping Connected to Reactor Pressure Vessel Nozzles

Condition	Load Combination for all Terms	Acceptance Criteria
Design	Design Pressure + Weight + Sustained Mechanical Load (Note 1)	In accordance with RPV and Nozzle Qualification Documentation/Specification
Service Levels A and B	Peak Pressure + Weight + Sustained Mechanical Load + Thermal Expansion (Note 1)	
Service Level B	Peak Pressure + Weight + AOO (Note 2) Where AOO can be the item as listed below: 1. AOO Plant/System Upset	
Service Level C	Peak Pressure + Weight + DBA Where DBA can be one of the items as listed below: 1. Emergency Level Transient 2. ATWS 3. SBLOCA	
Service Level D	Peak Pressure + Weight + [DBA or DEC] Where [DBA or DEC] can be one of the items as listed below: 1. SBLOCA + SSE 2. Large Break LOCA + SSE 3. Faulted Level Transient 3. CLE	

Notes:

(1) Effect of sustained mechanical load is considered together with pressure and weight. In normal operation load, fluid flow (including Flow-Induced Vibration when applicable), thermal and fluid reaction forces is considered together with pressure and weight.

(2) An event category assigned to a PIE or event sequence with an expected frequency of 1/year to 1.0 E02/year that represents a deviation from normal operation but does not evolve into a postulated accident because of DL2 functions.

Table 3A-23: Load Combinations and Acceptance Criteria for Pipe-Mounted Valve Acceleration

Condition	Load Combination for all Terms	Acceptance Criteria
Service Levels B	AOO (Note 1)	In accordance with applicable valve specification
Service Level C	DBA Where DBA can be one of the following items listed below: 1. Emergency Level Transient 2. ATWS 3. SBLOCA	
Service Level D	[DBA or DEC] Where [DBA or DEC] can be one of the following items: 1. SBLOCA + SSE 2. Large Break LOCA + SSE 3. Faulted Level Transient 3. CLE	

Note:

* * Notes under Table 3A-15 and Table 3A-16 apply to Table 3A-23.

(1) An event category assigned to a PIE or event sequence with an expected frequency of 1/year to 1.0 E02/year that represents a deviation from normal operation but does not evolve into a postulated accident because of DL2 functions.

Table 3A-24: Load Combinations and Acceptance Criteria for ASME B31.1 and ASME B31.3 Linear Type Supports (Anchor, Guide, Strut)

Description	Load Combination	Acceptance Criteria
Sustained	Design Pressure + Weight + Other Mechanical Loads	Vendor Load Capacity Datasheet, Vendor Design Report Summary, or in accordance with rules of ASME B31.1-2022 and ASME B31.3- 2022
Thermal and Sustained	Displacement Stress Range + Design Pressure + Weight + Other Mechanical Loads	
Occasional	Displacement Stress Range + Design Pressure + Weight + Other Mechanical Loads + Seismic Seismic event includes but is not limited to: SSE CLE	
Occasional	Displacement Stress Range + Design Pressure + Weight + Other Mechanical Loads + Occasional event other than Seismic Occasional event includes but is not limited to: AOO Plant/System Upset (Note 1) Emergency Level Transient ATWS SBLOCA Large Break LOCA Faulted Level Transient Wind Snow Load	

Note:

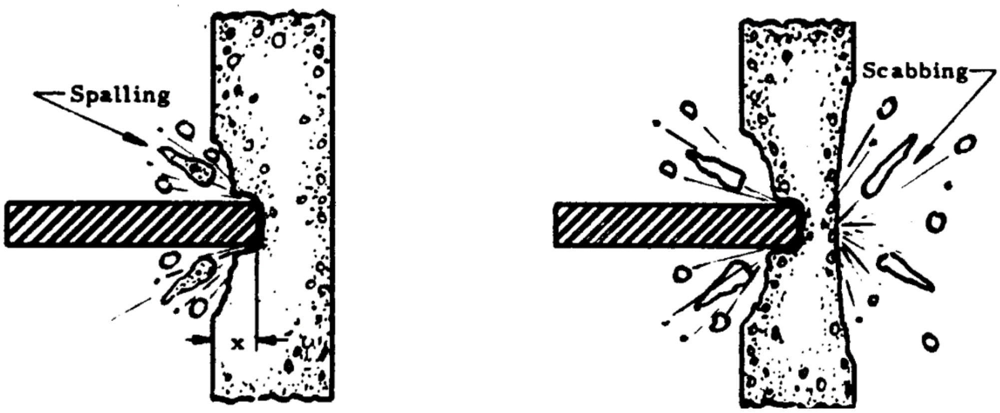
(1) An event category assigned to a PIE or event sequence with an expected frequency of 1/year to 1.0 E-02/year that represents a deviation from normal operation but does not evolve into a postulated accident because of DL2 functions.

Table 3A-25: ASME Section III Criteria for Selection and Testing of Bolting Materials

Code Category		ASME Class 1 Criteria	ASME Class 2 and 3 Criteria
Material Selection		NCA-1220 and NB-2128	NCD-2128
Material Test Coupons and Specimens for Ferritic Steel Material (Tensile Test Criteria)	Heat Treatment Criteria	NB-2210	NCD-2210
	Test Coupons requirements bolting and stud materials	NB-2224(b)	NCD-2224
Fracture Toughness Requirements	Materials to be impact tested	NB-2311 (a)(2)	NCD-2311
	Types of impact test	NB-2321	NCD-2321
	Location of Test Specimens Orientation of Impact Test Specimens	NB-2322.1 NB-2322.2(a)(3)	NCD-2322.1 NCD-2322.2
	Acceptance Standards	NB-2333	NCD-2332.3 NCD-2333.2
	Number of impact tests necessary	NB-2345	NCD-2345
	Retesting	NB-2350	NCD-2352
	Calibration of test equipment	NB-2360	NCD-2360
Examination criteria for bolts, studs, and nuts		NB-2541 NB-2580	NDC-2580
CMTR criteria		NB-2130 NCA-3860	NCD-2130 NCA-3860

Note:

* Section III paragraphs listed in this table represent those specified in the 2021 Edition of Section III.



(a) Missile Penetration and Spalling (b) Target Scabbing

Figure 3A-1: BWRX-300 Missile Impact Local Damage Modes for the Concrete Barrier

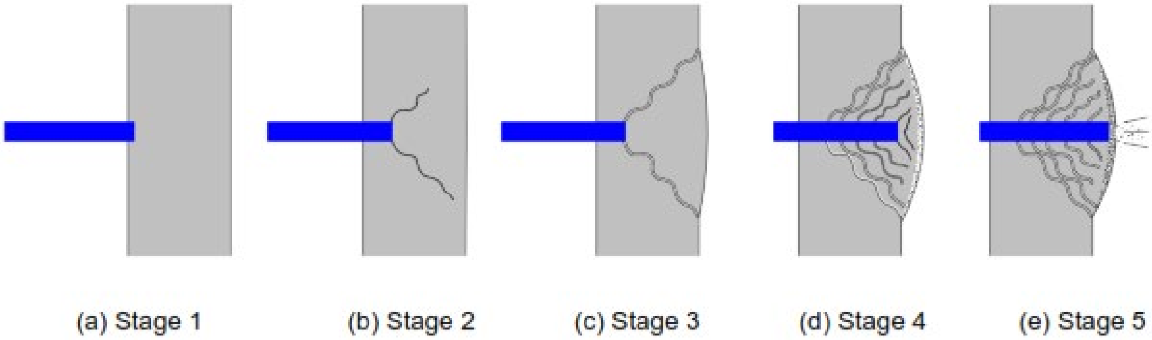


Figure 3A-2: BWRX-300 Missile Impact Local Damage Modes for the Steel-Plate Composite Barrier

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APPENDIX D COMPUTER PROGRAMS USED IN THE DESIGN AND ANALYSIS OF BWRX-300 SEISMIC CATEGORY STRUCTURES

D.1 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 3A-110), which takes cognizance of RGPs, such as ASME NQA-1, "Quality Assurance Requirements for Nuclear Facility Applications" (Reference 3A-139) and CSA N286.7-16, "Quality Assurance of Analytical, Scientific, and Design Computer Programs" (Reference 3A-140).

GEH maintains an ISO 9001:2015, "Quality Management Systems – Requirements", 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 reflect GEH's capabilities to meet the management system and high energy reactor facilities requirements described in IAEA Safety Standards Series – GSR Part 2: "Leadership and Management for Safety" (Reference 3A-141).

D.2 ACS SASSI v4

Description: ACS SASSI is a finite element computer code on the Microsoft Windows PC platforms for performing 3D dynamic SSI analysis to analyse 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 SSSI analyses, as applicable.

D.3 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 analysing 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.

D.4 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 analyse BWRX-300 structures for effects of blast loading and aircraft impact.

D.5 SSI-StressCoordTrans01P

Description: The SSI StressCoordTrans01P program is an auxiliary program to post-process the ACS SASSI NQA V4 STRESS result binary database. The SSI StressCoordTrans01P program includes an ensemble of STRESS database processing functionalities which was customized for application to the BWRX-300 SMR seismic SSI analysis projects. The SSI StressCoordTrans01P customized program is based on specific implementations incorporated in the ACS SASSI NQA V4 User Interface 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 SSI StressCoordTrans01P Program is used for post-processing the ACS SASSI STRESS binary databases for Integrated RB walls and floors in batch mode.

D.6 GT STRUDL

Description: GT STRUDL® is structural engineering software offering a complete design solution, including 3D Computer Aided Design (CAD) modelling and 64-bit high-performance computation solvers into all versions. GT STRUDL includes all the tools necessary to analyse a broad range of structural engineering and finite element analysis problems, including linear and non-linear static and dynamic 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: GT STRUDL is used for the structural analysis and design of non-BWRX-300 Seismic Category 1A structures.

APPENDIX E COMPUTER PROGRAMS USED IN THE DESIGN AND ANALYSIS OF MECHANICAL STRUCTURES, SYSTEMS, AND COMPONENTS

E.1 Introduction

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 3A-110), which takes cognizance of RGPs, such as ASME NQA-1 "Quality Assurance Requirements for Nuclear Facility Applications," (Reference 3A-139) and CSA N286.7-16, "Quality Assurance of Analytical, Scientific, and Design Computer Programs," (Reference 3A-140).

GEH maintains an ISO 9001:2015 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 reflect GEH's capabilities to meet the management system and high energy reactor facilities requirements described in IAEA Safety Standards Series – GSR Part 2: "The Management System for Facilities and Activities," (Reference 3A-141).

E.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 analysing 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.

E.3 PBLE v1

Description: Steam Dryer Analysis.

Validation: Validation of this tool is in compliance with the 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.

E.4 SIMCENTER 3D Acoustics v2022

Description: Used for modelling dryer acoustic loads and instrumentation diagnostics. Simcenter 3D is a unified, scalable, open, and extensible environment for 3D computer aided engineering 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.

E.5 GT STRUDL

Description: GT STRUDL® is structural engineering software offering a complete design solution, including 3D CAD modelling and 64-bit high-performance computation solvers into all versions. GT STRUDL includes all the tools necessary to analyse 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.

E.6 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.

E.7 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 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, MSIVs, SRVs, 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 and PISYS version 08P 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.

E.8 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-consistent 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 5% damping value at all frequencies recommended by ASME Non-Mandatory Appendix N to Section III (Reference 3A-127).

Validation: Hand calculation and test cases analysed are used to demonstrate the program's applicability and validity.

Extent of Application: This program is used to generate multiple damping spectra for piping.

E.9 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.

E.10 PIPESTRESS

Description: PIPESTRESS (developed under a Quality Assurance Program compliant with the ASME NQA-1 (Reference 3A-139) standard along with 10 CFR 21, "Reporting of Defects and Noncompliance," (Reference 3A-142) and 10 CFR 50 Appendix B, "Quality Assurance Criteria for Nuclear Power Plants and Fuel Reprocessing Plants," (Reference 3A-143) 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.

E.11 FLOMASTER v2021.1

Description: Uses simulation to offer reliable and 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.

E.12 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 analyse BWRX-300 structures for effects of blast loading and aircraft impact.

E.13 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.

APPENDIX F COMPUTER PROGRAMS USED IN THE DESIGN AND ANALYSIS OF ELECTRICAL STRUCTURES, SYSTEMS AND COMPONENTS

F.1 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 3A-110), which takes cognizance of RGPs, such as ASME NQA-1, "Quality Assurance Requirements for Nuclear Facility Applications" (Reference 3A-139) and CSA N286.7-16, "Quality Assurance of Analytical, Scientific, and Design Computer Programs" (Reference 3A-140).

GEH maintains an ISO 9001:2015, "Quality Management Systems – Requirements", 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 reflect GEH's capabilities to meet the management system and high energy reactor facilities requirements described in IAEA Safety Standards Series – GSR Part 2: "The Management System for Facilities and Activities" (Reference 3A-141).

F.2 ETAP v2021.1 (ICE Systems/I&C Tech)

Description: Electrical Transient Analyser Program (ETAP) is an electrical network modelling and simulation software tool used by power systems engineers to create an "electrical digital twin" and analyse 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.

F.3 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 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).

F.4 Quartus II (I&C Tech/ICE Systems)

Description: Tools that provide FPGA complier, simulation, and programming capabilities.

Validation: Validation of this tool is in compliance with the 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 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 G COMPUTER PROGRAMS USED IN THE DESIGN AND ANALYSES OF STRUCTURES, SYSTEMS, AND COMPONENTS – NUCLEAR FUELS

G.1 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 3A-110), which takes cognizance of RGPs, such as ASME NQA-1, "Quality Assurance Requirements for Nuclear Facility Applications" (Reference 3A-139) and CSA N286.7-16, "Quality Assurance of Analytical, Scientific, and Design Computer Programs" (Reference 3A-140).

GEH maintains an ISO 9001:2015 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 reflect GEH's capabilities to meet the management system and high energy reactor facilities requirements described in IAEA Safety Standards Series – GSR Part 2: "The Management System for Facilities and Activities" (Reference 3A-141).

G.2 EPRI: ACUBE v2

Description: Advanced cutset UB 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 analyses minimal cutsets and returns an estimate of the probability for a given top event using the Binary Decision Diagram (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.

G.3 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.

G.4 EPRI: MAAP v5

Description: The Modular Accident Analysis Program (MAAP) Version 5 - an Electric Power Research Institute (EPRI) owned and licensed 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 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 analyse 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.

G.5 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 can define 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.

G.6 FURST (Core and Fuel)

Description: Static and dynamic modelling.

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 internal loads, deflections, and stress analysis for BWRX-300.

G.7 GTRAC v1

Description: Post-processing TRACG graphics file to edit desired output.

Validation: Validation of this tool is in compliance with the 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.

G.8 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 USNRC. MACCS2 evaluates doses and health risks from the accidental atmospheric releases of radionuclides. 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.

G.9 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 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 k_{eff} eigenvalues for fissile systems is also a standard feature.

G.10 ORIGEN v2

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 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.

G.11 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 tools is in compliance with the project quality plan.

Extent of Application: The BWR Core Simulator (PANAC11A/P) is a steady-state, three-dimensional coupled nuclear-thermal hydraulic computer program representing a BWR core. An automated plant heat balance option is used for modelling 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, refuelling 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.

G.12 PRIME v3

Description: The PRIME03P computer program is used to calculate the thermal / mechanical response of nuclear fuel to time varying power histories.

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: 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.

G.13 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 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.

G.14 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.

G.15 RAMP: DandD v2.1

Description: A code for screening analyses for license 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 3A-144).

G.16 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 USNRC.

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 analyse hazards and design controls to prevent or mitigate potential accidents.

G.17 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.

G.18 RAMP: NRC-RADTRAN v6.02.1

Description: Risk and 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.

G.19 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.

G.20 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 Federal Monitoring and Assessment Code (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 Monitoring and Assessment Center (FRMAC).

G.21 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 to ensure compliance with skin dose limits.

G.22 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 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.

G.23 SCALE v6

Description: A Comprehensive Modelling 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.

G.24 TGBLA v6

Description: Calculates lattice parameters for fuel bundles and the output is used by PANACEA to model the behaviour of the fuel in the core.

Validation: Validation of this tool is in compliance with the 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 centred 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 analysing 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. LANCR will replace TGBLA.

G.25 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 modelling standard BWR fuels and advanced fuel designs including part length fuel rods and large water rods. In addition to modelling the BWR, TRACG04 is also applicable to experimental test facilities constructed from components representative of a BWR.

G.26 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 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.

G.27 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 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 3A-145) 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 or ANSI/ANS-5.1-1994, “American National Standard Decay Heat Power in Light Water Reactors," (Reference 3A-146) 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.

G.28 GTRAC v1

Description: Post-processing TRACG graphics file to edit desired output.

Validation: Validation of this tool is in compliance with the 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 H COMPUTER PROGRAMS USED IN ENVIRONMENTAL AND RADIOLOGICAL ANALYSES SUPPORTING THE DESIGN OF STRUCTURES, SYSTEMS, AND COMPONENTS

H.1 Introduction

This appendix describes the major computer programs used in environmental and radiological analyses supporting the design of SSCs. 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 3A-110), which takes cognizance of RGPs, such as ASME NQA-1, "Quality Assurance Requirements for Nuclear Facility Applications" (Reference 3A-139) and CSA N286.7-16, “Quality Assurance of Analytical, Scientific, and Design Computer Programs” (Reference 3A-140).

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- 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 reflect GEH’s capabilities to meet the management system and high energy reactor facilities requirements described in IAEA Safety Standards Series – GSR Part 2: “The Management System for Facilities and Activities” (Reference 3A-141).

H.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 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 3A-145) 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 or ANSI/ANS-5.1-1994 (Reference 3A-146) 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.

H.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.

H.4 RAMP: GALE v3.2

Description: The GALE series of codes consists of four codes that calculate the gaseous and liquid effluent releases from pressurized-water reactors (PWRs) and 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.

H.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.

H.6 RAMP: DandD v2.1

Description: A code for screening analyses for license 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 justification of parameter selections documented in Volumes 1 and 3 of NUREG/CR-5512 (Reference 3A-144).

H.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 analyse hazards and design controls to prevent or mitigate potential accidents.

H.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.

H.9 RAMP: NRC-RADTRAN v6.02.1

Description: Risk and 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.

H.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.

H.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:

- RDDs
- Nuclear Power Plant Emergencies
- Fuel Handling Accidents
- Transportation Accidents
- Nuclear Detonations

Turbo FRMAC calculations are based on methods established by the FRMAC.

H.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 to ensure compliance with the skin dose limits.

H.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 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.

H.14 SCALE v6

Description: A Comprehensive Modelling 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.

H.15 TGBLA v6

Description: Calculates lattice parameters for fuel bundles and the output is used by PANACEA to model the behaviour of the fuel in the core.

Validation: Validation of this tool is in compliance with the 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 centred 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 analysing 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. LANCR will replace TGBLA.

H.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 modelling standard BWR fuels and advanced fuel designs including part length fuel rods and large water rods. In addition to modelling the BWR, TRACG04 is also applicable to experimental test facilities constructed from components representative of a BWR.

H.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," (Reference 3A-147). The code calculates the doses from routine effluent emission from a plant that are the results of normal operation.

APPENDIX I COMPUTER PROGRAMS USED IN THE DESIGN IN THE DESIGN OF COMPONENTS, SYSTEMS, AND STRUCTURES IN SAFETY ANALYSIS (PROBABILISITIC AND DETERMINISTIC)

I.1 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 3A-110), which takes cognizance of RGPs, such as ASME NQA-1, "Quality Assurance Requirements for Nuclear Facility Applications" (Reference 3A-139) and CSA N286.7-16, "Quality Assurance of Analytical, Scientific, and Design Computer Programs" (Reference 3A-140).

GEH maintains an ISO 9001:2015 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 reflect GEH's capabilities to meet the management system and high energy reactor facilities requirements described in IAEA Safety Standards – GSR Part 2: "The Management System for Facilities and Activities" (Reference 3A-141).

I.2 EPRI: ACUBE v2

Description: Advanced cutset UB 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 analyses 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.

I.3 EPRI: CAFTA v11

Description: CAFTA is an integrated tool to perform Probabilistic Risk Analysis, incorporating 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.

I.4 EPRI: FRANX v4.4

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 analysing 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.

I.5 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 that 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.

I.6 EPRI: HRA Calculator v5.2

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.

I.7 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 ALWR designs. It can simulate Loss-Of-Coolant Accident (LOCA) and non-LOCA transients for PRA applications as well as severe accident sequences, including actions taken as part of the 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 analyse 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.

I.8 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.

I.9 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.

I.10 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.

I.11 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 perform 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.

I.12 EPRI: UNCERT v4

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.

I.13 GOTHIC v8

Description: The GOTHIC computer code is a state-of-the-art program for modelling multiphase, multicomponent fluid flow for performing both containment DBA analyses and analyses to support equipment qualification. The GOTHIC code is developed by Numerical Applications Incorporated, and the development program is sponsored by EPRI.

Validation: The software qualification process is being followed and verification and validation is in progress.

Extent of Application: GOTHIC is used to perform containment analysis.

I.14 RAMP: NRC-RADTRAN v6.02.1

Description: Risk and 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.

I.15 SCALE v6

Description: A Comprehensive Modelling 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.

I.16 TRACG v4

Description: TRACG is a GEH version of the Transient Reactor Analysis Code which is 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-DW 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 modelling standard BWR fuels and advanced fuel designs including part length fuel rods and large water rods. In addition to modelling the BWR, TRACG04 is also applicable to experimental test facilities constructed from components representative of a BWR.

I.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.