GE Hitachi Nuclear Energy

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BWRX-300 UK Generic Design Assessment (GDA)
BWRX-300 Chapter 9B – Civil Structures

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EXECUTIVE SUMMARY

This Preliminary Safety Report (PSR) chapter describes how the design of specific structures in the BWRX-300 plant complies with the general design requirements identified in PSR Ch. 3.

Chapter 9B considers three groups of structures: Foundations, the Reactor Building (RB) and Other Structures. The Other Structures are the Radwaste Building (RWB), the Control Building (CB), Turbine Building (TB), the Service Building, the Reactor Auxiliary Structures and the Fire Water storage Tank and Pump Enclosure.

The civil engineering structures are described with the use of General Arrangement drawings showing both plan and sectional views of the power block structures and their foundations. The chapter defines the structural role and safety function of each structure and how they will be designed and constructed. Materials used and interfaces with other Structures, Systems and Components (SSCs) are outlined. The range of geotechnical conditions, the hazards and loading conditions required to satisfy the safety demands of the BWRX-300 Standard Plant design are discussed. The maintenance and aging management of structures are also outlined.

The features of the Steel-Plate Composite Containment Vessel (SCCV) are presented to demonstrate the performance of the containment in all plant states and to justify containment robustness under Design Basis and Design Extension Conditions.

The design maturity of the Service Building, Reactor Auxiliary Structures and Fire Water storage Tank and Pump Enclosure limits the information available for this revision of PSR Ch. 9B. Nonetheless a high-level summary of the safety approach is provided to ensure that the relevant safety standards will be met as their designs progress.

A Claims-Arguments-Evidence Route Map is laid out in Appendix A. A Forward Action Plan listing additional items to be addressed within the Generic Design Assessment of the BWRX-300 is detailed in Appendix B.

ACRONYMS AND ABBREVIATIONS

Acronym	Explanation
ABWR	Advanced Boiling Water Reactor
ACI	American Concrete Institute
AISC	American Institute of Steel Construction
ALARP	As Low As Reasonably Practicable
ALWR	Advanced Light-water Reactor
ANSI	American National Standard Institute
AOO	Anticipated Operational Occurrence
ASCE	American Society of Civil Engineers
ASME	American Society of Mechanical Engineers
BDBA	Beyond Design Basis Accident
BDBE	Beyond Design Basis Earthquake
BE	Best Estimate
BPVC	Boiler and Pressure Vessel Code
BWR	Boiling Water Reactor
CAE	Claims Argument Evidence
CB	Control Building
CDM	Construction (Design and Management)
CEPSS	Containment Equipment and Piping Support Structure
DBA	Design Basis Accident
DBE	Design Basis Earthquake
DCIS	Distributed Control and Information System
DEC	Design Extension Condition
D-in-D	Defense-in-Depth
DP-SC	Diaphragm Plate Steel-Plate Composite
EPRI	Electrical Power Research Institute
FAP	Forward Action Plan
FE	Finite Element
FIA	Foundation Interface Analysis
GDA	Generic Design Assessment
GDC	General Design Criteria
GDRS	Generic Design Response Spectra
GEH	GE Hitachi Nuclear Energy
HCLPF	High Confidence of Low Probability of Failure
HELB	High Energy Line Break
HVAC	Heating, Ventilation, and Air Conditioning
IAEA	International Atomic Energy Agency

Acronym	Explanation
ILRT	Integrated Leak Rate Testing
ISI	Inservice Inspection
ISRS	In-Structure Response Spectra
ksi	Thousands of pounds per square inch
LB	Lower Bound
LfE	Learning from Experience
LOCA	Loss of Coolant Accident
LWM	Liquid Waste Management System
MCR	Main Control Room
NRIC	National Reactor Innovation Center
NS	Non-Seismic
OBE	Operating-Basis Earthquake
OGS	Offgas System
ONR	Office for Nuclear Regulation
OPEX	Operating Experience
PIE	Postulated Initiating Event
PSR	Preliminary Safety Report
RB	Reactor Building
RGP	Relevant Good Practice
RPV	Reactor Pressure Vessel
RWB	Radwaste Building
SC	Safety Class
SCCV	Steel-Plate Composite Containment Vessel
SCDS	Safety Case Development Strategy
SCR	Secondary Control Room
SEI	Structural Engineering Institute
SFR	Safety Functional Requirement
SIT	Structural Integrity Test
SMR	Small Modular Reactor
SRP	Standard Review Plan
SSCs	Structures, Systems and Components
SSI	Soil-Structure Interaction
SSSI	Structure-Soil-Structure Interaction
SWM	Solid Waste Management System
TB	Turbine Building
UB	Upper Bound
U.S.	United States

Acronym	Explanation
USNRC	U.S. Nuclear Regulatory Commission
UK	United Kingdom
UPS	Uninterruptible Power Supply

SYMBOLS AND DEFINITIONS

Symbol	Definition
K_0	at-rest lateral coefficient
E_{st}	static elastic modulus
ν_{st}	static elastic Poisson's ratio

Term	Definition
Standard Design	The BWRX-300 design uses a Standard Design approach to minimize the variation from project to project.
Reactor Building	Reactor Building (RB) is used to refer to the part of the integrated structure located outside of containment.
Integrated Reactor Building	The integrated RB structure is inclusive of the RB, containment, and containment internal structures.
One-Step Approach	Response analysis for both horizontal and vertical components of motion performed by the one-step method as defined in Section 3.1.2 of ASCE/SEI 4-16.
Power Block	The primary buildings in the BWRX-300 Power Block consist of the RB which houses the containment, Radwaste Building (RWB), Control Building (CB), Turbine Building (TB), Service Building and Reactor Auxiliary Structures.
Severe Environmental Loads	Loads to be sustained during severe environmental conditions, including those induced by the design wind and the Operating-Basis Earthquake (OBE) specified for the plant site during expected plant operational states.
Extreme Environmental Loads	Loads to be sustained during extreme environmental conditions, including those induced by the extreme wind and the Design Basis Earthquake (DBE) specified for the plant site.

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REVISION SUMMARY

Revision #	Section Modified	Revision Summary
A	All	Initial Issuance

9B CIVIL STRUCTURES

Preliminary Safety Report (PSR) Ch. 9B describes how the general design requirements specified in PSR Ch. 3, NEDC-34165P, “Safety Objectives and Design Rules for SSCs,” (Reference 9B-1), Section 3.5: Design of Civil Structures are complied with in the design of civil structures.

Section 9B.1 focuses on the buried structures and foundations supporting the Power Block structures and the Fire Water Storage Tank and Pump Enclosure located outside of the Power Block.

The Power Block structures consist of the Reactor Building (RB) housing the containment, the Radwaste Building (RWB), Control Building (CB), Turbine Building (TB), Service Building and Reactor Auxiliary structures.

Section 9B.2 focuses on each of the RB integrated structures. As mentioned in PSR Ch. 3, Section 3.3.1, reference to the integrated RB is inclusive of the RB, containment, and containment internal structures, whereas RB is used to refer to the RB structure outside of containment.

Section 9B.3 provides the general design requirements for the RWB, CB, TB, Service Building, Reactor Auxiliary Structures and Fire Water Storage Tank and Pump Enclosure. The information presented for these structures in this chapter is commensurate with their importance to safety and their level of maturity. The remaining plant structures are not covered because they are not credited in the safety analysis or are of a site-specific nature, as is the case for the Pumphouse and Forebay Structures.

Below are the key PSR chapters that should be reviewed along with this chapter:

1. PSR Ch. 1 – NEDC-34163P, “BWRX-300 UK GDA Ch. 1: Introduction and General Considerations,” (Reference 9B-2), 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.
2. PSR Ch. 2 – NEDC-34164P, “BWRX-UK GDA Ch. 2: Site Characteristics,” (Reference 9B-3), 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 UK.
3. PSR Ch. 3 – Section 3.1, NEDC-34165P, which provides the general design aspects and Defense-in-Depth (D-in-D) safety framework utilized in the BWRX-300 design.
4. PSR Ch. 3 – Section 3.2, NEDC-34165P, which provides the general classification of BWRX-300 SSCs and the approach used to establish these classifications.
5. PSR Ch. 3 – Sections 3.3 and 3.4, NEDC-34165P, which provide methodology and general design requirements for protection against the effects of external and internal hazards.
6. PSR Ch. 6 – NEDC-34168P, “BWRX-300 UK GDA Ch. 6: Engineered Safety Features,” (Reference 9B-4), which describes the containment and its associated systems, engineered safety features included to mitigate the consequences of anticipated operational occurrences or postulated design basis accidents, and habitability systems.
7. PSR Ch. 9A – NEDC-34171P, “BWRX-300 UK GDA Ch. 9A: Auxiliary Systems,” (Reference 9B-5), which discusses BWRX-300 auxiliary systems including the

Heating, Ventilation, and Air-Conditioning (HVAC), the Fire Protection systems, overhead lifting equipment and floor drain systems.

8. PSR Ch. 12 – NEDC-34175P, “BWRX-300 UK GDA Ch. 12: Radiation Protection,” (Reference 9B-6), which describes the administrative programs and procedures, in conjunction with facility design, that ensure occupational radiation exposure to personnel is kept As Low As Reasonably Practicable (ALARP).
9. PSR Ch. 15 – NEDC-34178P, “BWRX-300 UK GDA Ch. 15: Safety Analyses,” (Reference 9B-7), which discusses Design Extension Conditions (DECs) and Beyond Design Basis Accidents (BDBAs) considered in the design and documents safety analysis results used to confirm adequacy of design.
10. PSR Ch. 25 – NEDC-34917P, “BWRX-300 UK GDA Ch. 26 Security Annex,” (Reference 9B-8), which evaluates the ability of the integrated RB to withstand regulatory defined threats.

A Claims-Arguments-Evidence Route Map is laid out in Appendix A. A Forward Action Plan (FAP) listing additional items to be addressed within the Generic Design Assessment of the BWRX-300 is detailed in Appendix B.

9B.1 Foundations and Buried Structures

The BWRX-300 containment and RB, including their foundation, are constructed using steel-plate composite modules with diaphragm plate referred as Diaphragm Plate Steel-Plate Composite (DP-SC) modules. The diaphragm plates of DP-SC modules have holes to allow the flow of concrete during construction and are used along with headed studs steel anchors to provide the composite action between the steel faceplates and the concrete core.

The foundations of all other structures are constructed using standard reinforced concrete and are assumed to be near surface raft slab foundations, however the foundation characteristics may vary depending on selected site conditions.

This section discusses the design basis of each of these foundation types.

9B.1.1 Integrated RB Foundation

9B.1.1.1 Structural Role and Safety Function

9B.1.1.1.1 Structural Role

The common DP-SC mat foundation supports the deeply embedded integrated RB structure and provides the mechanism for transferring all applicable loads from the integrated RB structure to the surrounding subgrade.

9B.1.1.1.2 Safety Design Bases

Detailed Safety Functional Requirements (SFRs) for the Integrated RB foundation will be developed later when more detail is available for the Pre-Construction Safety Report. The primary function of foundation structure is to support SSCs, provide stability, prevent differential displacement between adjacent structures and act as a barrier between the internal building and the external environments.

The integrated RB common mat foundation is classified as Safety Class 1 (SC1) consistent with the highest-class component housed in the building.

The seismic category of the containment and RB common mat foundation is the same as that of the superstructures (BWRX-300 Seismic Category 1A, per PSR Ch. 3, Table 3-1). Conforming with the regulatory guidance of Section I.3 of NUREG-0800, Standard Review Plan (SRP) 3.8.5, “Foundations,” (Reference 9B-9), the common mat foundation is designed to ensure the stability and structural integrity of the integrated RB under normal operation, Anticipated Operational Occurrences (AOOs) and Design Basis Accidents (DBAs) and evaluated for DECs.

A portion of the integrated RB is a deeply embedded structure, with approximately 34.0 m residing below grade. NEDO-33914-A, “BWRX-300 Advanced Civil Construction and Design Approach,” (Reference 9B-10) provides guidelines for the development of site-specific geotechnical and seismic design parameters to ensure appropriate soil and rock parameters, and subgrade modeling assumptions, are established for the seismic Soil-Structure Interaction (SSI) of the deeply embedded integrated RB, including foundation, at candidate sites. The BWRX-300 Standard Design (as discussed in PSR Ch. 3, Section 3.3.1.1) considers generic site parameters bounding a wide range of conditions existing at candidate sites across North America following the guidelines in NEDO-33914-A. The Standard Design subgrade parameters used in the analysis and design of the integrated RB are presented in Section 7.0 of NEDO-33914-A. Subgrade modeling assumptions for the SSI of the deeply embedded integrated RB are described in Section 9B.2.5.

Stability evaluation of the common mat foundation is performed using the criteria in Section 9B.1.1.3.2. A potential liquefiable evaluation of the soil strata is performed, and the effects of fluctuating groundwater are considered, in the evaluation of soil capacity. This is done to preclude total and differential settlement of the common mat foundation that can challenge the functions of BWRX-300 Seismic Category 1A or 1B SSCs in the integrated RB.

9B.1.1.2 Structural Description

The deeply embedded cylindrical integrated RB structure is founded on a common circular mat foundation that supports the containment, containment internal structures and RB. This common mat foundation is made of DP-SC modules, the same as the supported superstructures, and is deeply embedded in the subgrade. The containment boundary for the common mat foundation extends to the outer perimeter of the Steel-Plate Composite Containment Vessel (SCCV) as shown in Figure 9B-1. The mat foundation supporting the SCCV is referred to as the inner mat foundation (see Figure 9B-2) and is part of the SCCV. The mat foundation outside of the containment boundary is referred to as the outer mat foundation.

The walls of the RB and containment structures carry the vertical loads from the structures to the deeply embedded common mat foundation. Lateral loads are transferred to these walls and to wing walls by the roof, floor diaphragms and Containment Equipment and Piping Support Structure (CEPSS). The below grade RB exterior wall and mat foundation transmit, in turn, the loads to the surrounding subgrade. The transfer of the lateral loads is affected by the interaction of the deeply embedded integrated RB structure with the surrounding soil and rock materials as discussed in Section 9B.2.5.

Figure 9B-1 and Figure 9B-2 provide dimensions, plan, and section views of the integrated RB common foundation.

9B.1.1.3 Structural Analysis and Design Basis

9B.1.1.3.1 Applicable Codes, Standards and Other Specifications

Applicable codes, standards and specifications for the containment and RB common DP-SC foundation are the same as those for the superstructures described in Section 9B.2.

The jurisdictional boundary for the application of Section 6.0 of NEDC-33926P, “Steel-Plate Composite Containment Vessel and Reactor Building Structural Design,” (Reference 9B-11) to the containment is the portion within the perimeter or exterior surface of the SCCV as shown in Figure 9B-3.

The jurisdictional boundary for application of the American National Standard Institute (ANSI)/American Institute of Steel Construction (AISC) N690, “Specification for

Safety-Related Steel Structures for Nuclear Facilities," (Reference 9B-12) and the modified design rules in Section 5.0 of NEDC-33926P to the non-pressure retaining portion of the common foundation is the portion spanning from the exterior surface of the SCCV to the exterior surface of the RB (see Figure 9B-3).

9B.1.1.3.2 Bounding Subgrade Design Parameters

Following the guidance of Section 7 of NEDO-33914-A, the BWRX-300 standard design considers bounding subgrade design parameters for a set of generic seismological and geotechnical site properties representing a wide range of types and conditions existing at candidate sites across North America.

Three sets of Generic Design Response Spectra (GDRS) define the horizontal and vertical components of the seismic design ground motion at sites with firm, medium, and hard subgrade stiffness properties. Eight generic profiles of dynamic and static subgrade properties provide a bounding representation of the various geotechnical conditions existing at the candidate sites for construction of the BWRX-300. The selected bounding subgrade and seismic parameters build confidence in the suitability of the BWRX-300 Standard Design for the majority of potential United Kingdom (UK) sites.

The Standard Design uses ground water pressure demands based on a conservatively selected nominal groundwater level located at plant grade. The same groundwater level is used in the stability calculations to account for the buoyancy force.

The bearing pressure calculations and construction optimization evaluations are performed considering two bounding groundwater level elevations located at plant grade and the underside of BWRX-300 RB foundation.

The exterior RB wall is the main structural member resisting the below grade lateral pressures applied on the RB integrated structures. These below grade lateral loads include the static earth pressure, ground water hydrostatic pressure, and additional rock pressure.

9B.1.1.3.2.1 Bounding Equivalent Linear Subgrade Static Profiles

As described in Section 9B.2.5, the structural design demands due to static earth pressures on the RB below grade exterior wall are obtained from the 1-g static analyses of the integrated RB Finite Element (FE) model embedded in a layered half-space continuum model representing the surrounding soil and rock. In support of the seismic SSI analyses, impedance static analyses are performed to evaluate the frequency-dependent stiffness properties of the layered subgrade materials. To account for the interaction of the RB integrated structures with the surrounding subgrade, super-elements representing the stiffness properties of the layered subgrade materials are used in the static and thermal analyses, as described in Section 9B.2.5.

The 1-g static SSI analyses, subgrade impedance analyses and thermal stress analyses use profiles of bounding equivalent linear soil and rock properties developed following the guidance of NEDO-33914-A, Section 5.2.1. They consist of:

- Effective unit weight that for soil materials below groundwater table are calculated as the total unit weight of soil minus the unit weight of water
- Elastic and Shear Modulus representing linearized stiffness properties of the soil and rock for long-term static loading conditions
- Soil and rock Poisson ratios representative of at-rest lateral pressure conditions. In accordance with the guidance of NEDO-33914-A, Section 5.2.1.1, the soil Poisson ratios (ν_{st}) are calculated as follows using the at-rest lateral (K_0) coefficient values

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

The Upper Bound (UB) unit weight, Lower Bound (LB) static elastic modulus (E_{st}) and UB coefficient of at-rest pressure (K_0) are used for the 1-g SSI analyses resulting in larger deformation at soil-structure interfaces and conservative design stress demands.

UB static elastic modulus (E_{st}) and UB Poisson ratios (ν_{st}) are used for thermal stress resulting in conservative thermal stress demands.

LB static elastic modulus (E_{st}) and LB Poisson ratios (ν_{st}) are used for the static impedance analyses.

9B.1.1.3.2.2 Soil Bearing Stability

The static and dynamic stability of soil supporting the integrated RB mat foundation is evaluated per the regulatory guidance of NUREG-0800, SRP 2.5.4, “Stability of Subsurface Materials and Foundations,” (Reference 9B-13), Section 2.5.4.10.

The bearing capacity of the soil materials supporting the foundations surrounding the RB is taken into consideration following the guidelines in NEDO-33914-A, Section 6.2.

The calculation of the dynamic bearing pressure demands under Design Basis Earthquake (DBE) loads from the results of the seismic SSI analyses is discussed in PSR Ch. 3, Section 3.3.1.3.

Per Article 4.35 of International Atomic Energy Agency (IAEA) Safety Guide No. NS-G-3.6, “Geotechnical Aspects of Site Evaluation and Foundations for Nuclear Power Plants,” (Reference 9B-14), safety factors against potential bearing capacity failure of the subsurface materials depend on the method of bearing capacity evaluation and site conditions. If a conventional bearing capacity method is used, safety factors are not less than 3 under static loads.

9B.1.1.3.2.3 Foundation Stability

The integrated RB mat foundation stability is assessed against sliding and overturning due to DBE and extreme wind, and flotation following the regulatory guidance of NUREG-0800, SRP 3.8.5, Section II.4.

For the RB foundation stability against flotation, the design basis flood considers water level at grade.

Explicit sliding and overturning stability evaluations are not performed for the deeply embedded RB because, in accordance with Sections 7.2.1 and 7.2.2 of American Society of Civil Engineers (ASCE)/ Structural Engineering Institute (SEI) 43, “Seismic Design Criteria or Structures, Systems, and Components in Nuclear Facilities,” (Reference 9B-15), its centre of gravity is located below the grade elevation, and the structure is inherently stable against sliding and overturning. The foundation stability of the surrounding RWB, CB, TB, Service Building and Reactor Auxiliary Structures is checked to ensure that there is no adverse interaction with the RB during a DBE level event. Stability of the foundations surrounding the RB under DBE loads is evaluated using the results of the seismic SSI analyses as described in PSR Ch. 3, Section 3.3.1.3.

Safety factors against sliding and overturning for the Power Block foundations under normal operating and accidental conditions are presented in Table 9B-1. Safety factors listed in Table 9B-1 conform to the allowable values specified in NUREG-0800, SRP 3.8.5, Section II.5.

9B.1.1.3.3 Loads and Load Combinations

9B.1.1.3.3.1 Design Loads

Design loads of the containment and RB common mat foundation are those of the superstructures described in Subsections 9B.2.1.3.2 and 9B.2.3.3.2.

9B.1.1.3.3.2 Design Load Combinations

Design load combinations of the containment and RB common mat foundation are those of the superstructures described in Subsections 9B.2.1.3.2 and 9B.2.3.3.2.

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

9B.1.1.3.4 Design and Analysis Procedures

The design and analysis procedures for the DP-SC common mat foundation conform to the regulatory guidance in Section II.4 of NUREG-0800, SRP 3.8.5.

As outlined in Section 9B.2.5, the common mat foundation supports the integrated RB structures and provides the mechanism for transferring all applicable loads from the integrated RB to the surrounding subgrade.

The containment and RB common mat foundation is analyzed using the methods where the transfer of loads from the foundation mat to the supporting foundation media is determined by elastic methods. Demands for the design of the common mat foundation are obtained from the structural analyses described in Section 9B.2.5 performed on the integrated RB structural model that include the effects of interaction of the structure with the surrounding subgrade and the effects of the foundations of the surrounding Power Block buildings.

The common DP-SC foundation mat is represented by thick shell elements in the integrated FE model. Properties assigned to the shell elements representing the common DP-SC foundation in the dynamic FE model used for the seismic SSI analyses are described in PSR Ch. 3, Section 3.3.1.3. Properties assigned to the foundation shell elements in the integrated FE models used for the static and thermal stress analyses are described in Section 9B.2.5. Subgrade properties used for the seismic SSI analysis of the mat foundation are discussed in PSR Ch. 3, Section 3.3.1.2. Subgrade properties used for the static and thermal stress analyses of the mat foundation are discussed in Section 9B.1.1.3.2.1.

The containment foundation is designed in accordance with Section 6.0 of NEDC-33926P consistent with U.S. Nuclear Regulatory Commission (USNRC) RG 1.136, "Design Limits, Loading Combinations, Materials, Construction, and Testing of Concrete Containments," (Reference 9B-16). The non-pressure retaining portion of the common foundation mat is designed to ANSI/AISC N690, supplemented by USNRC RG 1.243, "Safety-Related Steel Structures and Steel-Plate Composite Walls for Other Than Reactor Vessels and Containments," (Reference 9B-17) and NEDC-33926P, Section 5.0.

Effects of normal and differential settlement of BWRX-300 structures is considered in the design and include consideration of the effects of fluctuating ground water on the foundations per NUREG-0800 SRP 3.8.5.

The methodology used to validate the subgrade pressure loads on the common mat foundation is described in NEDO-33914-A, Section 5.1.3.

9B.1.1.3.5 Foundation Design Criteria

The structural acceptance criteria for the containment and RB common foundation are the same as those for their respective superstructures. Refer to Section 9B.1.1.3.2 for safety factors considered for soil bearing and foundations stability.

9B.1.1.3.6 Fire Protection

The design fire resistance and protection features of the containment and RB common mat foundation are those of the superstructures described in Subsections 9B.2.1.3.6 and 9B.2.3.3.5.

9B.1.1.4 Materials

Refer to Subsections 9B.2.1.4 and 9B.2.3.4 for a description of materials used for the common mat foundation supporting the BWRX-300 integrated RB.

9B.1.1.5 Interfaces With Other Structures, Systems or Components

The behavior of contact interfaces between the common circular mat foundation supporting the integrated RB structures and the surrounding supporting media is considered in the SSI analyses discussed in Section 9B.2.5. These analyses consider Structure-Soil-Structure Interaction (SSSI) due to the closeness of the RWB, CB, TB, Service Building and Reactor Auxiliary Structures to the RB as described in Section 9B.2.5.

To protect the mat foundation from damage following core melt, a corium shield (see Figure 9B-1) is installed at the lower portion of the SCCV, and Reactor Pressure Vessel (RPV) pedestal as described in Section 9B.2.6.1.

9B.1.1.6 System And Equipment Operation

Not applicable to structural design of the common mat foundation.

9B.1.1.7 High level construction considerations

Refer to NEDO-33914-A, Section 1.4 for the preferred construction approach for the deeply embedded RB.

Construction requirements are defined on drawings and/or specifications. Construction planning and constructability reviews are performed concurrently with the structural design.

Design development and construction method adopted for the deployment of the BWRX-300 plant takes into consideration Construction (Design and Management) (CDM) 2015 Regulations as described in PSR Ch. 14.

9B.1.1.8 Instrumentation And Control

As described in PSR Ch. 3, Section 3.3.1.5, seismic instrumentation is installed inside the RB at the top of the common mat foundation in accordance with USNRC RG 1.12, “Nuclear Power Plant Instrumentation For Earthquakes,” (Reference 9B-18) to monitor the seismic motions that may occur for the lifecycle of the reactor facility, starting from commissioning to decommissioning.

Piezometers, settlement sensors and extensometers are also used as described in Section 9B.1.1.9 to monitor the magnitude and distribution of pore pressure and amount of deformation of the subgrade materials surrounding the integrated RB.

9B.1.1.9 Monitoring, Testing, Inspection and Maintenance

Complying with the requirements of 10 CFR 50.65 and the regulatory guidance of USNRC RG 1.160, “Monitoring the Effectiveness of Maintenance at Nuclear Power Plants,” (Reference 9B-19), monitoring, testing, inspection, and maintenance programs are implemented in the BWRX-300 design to detect settling and reduce/repair degradation of the common mat foundation.

As mentioned in Section 9B.1.1.8, field instrumentation is installed inside and outside of the RB shaft to measure the distribution of pore pressures around and below the RB shaft, total settlement and tilt of the RB shaft including foundation, and measurement of the surrounding structures following the recommendations in NEDO-33914-A, Section 3.4. The instrumentation provides recordings during excavation and continuing through the BWRX-300 plant operation that can be benchmarked against design estimates using results of Foundation Interface Analysis (FIA) as described in NEDO-33914-A, Section 4.3.4. Short-term and long-term settlement monitoring plans are also developed to detect both vertical and horizontal movements in and around the integrated RB, as well as differential distortion across the foundation footprint and differential settlements between the containment and RB portions of the common mat foundations.

Refer to Subsections 9B.2.1.4.7 and 9B.2.1.9 for the construction, monitoring, testing, inspection, and maintenance requirements applicable to the containment portion of the mat foundation and to Subsections 9B.2.3.4.8 and 9B.2.3.9 for the requirements applicable to the non-containment portion of the mat foundation.

9B.1.1.10 Radiological Aspects

As discussed in Section 9B.1.1.2, the RB structure is founded on a common circular mat foundation consisting of an inner and outside containment boundary. For radiological aspects relating to the inner boundary refer to Section 9B.2.1.10, and outside containment refer to Section 9B.2.3.10.

9B.1.1.11 Performance And Safety Evaluation

To meet its functional and performance requirements listed in Section 9B.1.1.1, the mat foundation is evaluated and designed for a comprehensive set of design loads described in Section 9B.1.1.3.3 and combinations (see Table 9B-2 and Table 9B-3) using the method and design basis outlined in Section 9B.1.1.3 , with due consideration given to the probability of concurrence, loading time history and sequence of loads, as applicable.

For all design basis load combinations, the containment portion of the mat foundation meets the design requirements of Section 6.0 of NEDC-33926P.The non-containment portion of the mat foundation meets the design requirements of ANSI/AISC N690 and the modified rules in Section 5.0 of NEDC-33926P.

Stability evaluations for the common mat foundation from various loading arising throughout the plant life, including external hazards specified in PSR Ch. 3, Section 3.3, are performed using FIA results as described in NEDO-33914-A, Section 4.3.4. Allowable bearing capacities for the common mat foundation taking into account the dry and submerged conditions are considered. Stability and bearing capacity evaluations are performed for the integrated RB discussed in Section 9B.2 to ensure that the factors of safety listed in Table 9B-1 are met.

9B.1.2 Other Structure Foundations

The RWB, CB, TB, Service Building, Reactor Auxiliary Structures, and prefabricated Equipment Structures to the CB and TB are assumed to be supported on near surface reinforced concrete mat foundations, however the foundation characteristics may vary depending on selected site conditions. The mat foundations are separated from the RB exterior wall with a gap that can accommodate structural displacements due to seismic and extreme wind loads.

Due to the low maturity of the design of the Fire Water Storage Tank and Pump Enclosure, a description of the enclosure foundation and its design approach will be provided later, when more detail is available in the Pre-Construction Safety Report.

9B.1.2.1 Safety Design Bases

Detailed SFRs for the foundation structures described in Section 9B.1.2 will be developed later, together with each building’s SFRs, when more detail is available for the Pre-Construction Safety Report.

The primary function of each of the foundation structures is to support SSCs, provide stability, prevent differential displacement between adjacent structures and act as a barrier between the internal building and the external environments.

9B.1.2.2 Design Approach

The Power Block structures foundations are assigned the same safety classification and seismic category as the superstructures they support. The analysis and design of the Seismic Category RW RWB mat foundation is in accordance with American Concrete Institute (ACI) 349, “Code Requirements for Nuclear Safety-Related Concrete Structures and Commentary,” (Reference 9B-20). It is considered that, for structural design and analysis of the CB, TB, Service Building, Reactor Auxiliary Structures and Equipment Structures foundations, Eurocodes are used. Loads, load combinations and acceptance criteria for the foundations design are the same as those used for the design of their superstructures.

Stability evaluations of the Power Block structures foundations against sliding and overturning due to DBE, and extreme wind, and against floatation due to flooding, are performed to ensure the stability requirements under normal operating and accidental conditions, discussed in Section 9B.1.1.3, are met. Stability evaluations are performed using the results of the seismic SSI analyses discussed in PSR Ch. 3 Section 3.3.1. The liquefaction potential of the Power Block structures’ foundations is also assessed in line with NEDO-33914-A.

9B.2 Integrated Reactor Building

The integrated RB structure consists of the containment, containment internal structures and RB. The containment structure presented in Section 9B.2.1 protects and acts as a leak-tight boundary for the containment and associated systems discussed in PSR Ch. 6, Section 6.5. Internal structures housed in the containment are described in Section 9B.2.2 and include the pedestal that supports the RPV and structures supporting piping and equipment within the containment.

The containment structure is completely enclosed within the deeply embedded RB structure presented in Section 9B.2.3. The RB is designed to protect the containment structure from external hazards (e.g., wind loads, extreme wind loads (tornado and hurricane), aircraft hazard, missiles) and external beyond design basis scenarios. In addition to sharing the common mat foundation, the RB and containment structures are integrated at the connections of wing walls and elevated slabs, between the mat foundation, and the reactor cavity pool top slab elevation of the structure.

9B.2.1 Containment

The BWRX-300 uses a traditional containment system for the ultimate containment of radioactive materials for various postulated events. The configuration of the containment is shown in Figure 9B-1. It comprises a steel-plate composite containment vessel referred to as the SCCV, a steel containment closure head and other Class MC components. As shown in Figure 9B-1, the BWRX-300 containment is completely enclosed within the RB.

9B.2.1.1 Structural Role and Safety Function

9B.2.1.1.1 Structural Role

The primary functions of the containment structure are:

- To enclose and support the RPV and its connected piping systems
- To act as a leak-tight pressure boundary confining radioactive substances in different plant states, including normal operation, AOOs, DBAs, and DECs

- To provide radiation shielding to limit radiation dose within the applicable regulatory standards in different plant states, including normal operation, AOOs, DBAs, and DECs
- To provide in conjunction with the surrounding RB, protection for plant safety equipment and systems against external hazards and human induced events described in PSR Ch. 3, Section 3.3

9B.2.1.1.2 Safety Design Bases

Detailed SFRs for the SCCV will be developed later when more detail is available for the Pre-Construction Safety Report.

The BWRX-300 containment structure is classified as SC1 consistent with the highest-class component housed in the structure and is a BWRX-300 Seismic Category 1A structure as indicated in PSR Ch. 3, Table 3-1.

As mentioned in Section 9B.2.1.1.1, the main function of the containment structure is to enclose and support containment systems and piping, and to act as a leak-tight pressure boundary confining radioactive substances during normal operation, AOOs, DBAs, and DECs in compliance with the requirements of 10 CFR 50, Appendix A General Design Criteria (GDCs) 16 and 50.

To perform its intended safety functions, the containment structure is designed to resist potential internal flooding, over-pressures, under-pressures, temperatures, and dynamic effects as a result of internal missile generation and high energy line breaks discussed in PSR Ch. 3, Section 3.4, with sufficient margins of safety in compliance with requirements of 10 CFR 50 Appendix A GDC 4. Being enclosed within the RB, the containment is also protected from the external hazards discussed in PSR Ch. 3, Section 3.3 in compliance with the requirements of 10 CFR 50 GDC 2.

The design bases and the various modes of operation of safety features and systems used to ensure the confinement functions and leak tightness of the containment are described in PSR Ch. 6, Section 6.5.

For the beyond design basis robustness of the containment against overpressure, combustible gas pressures, seismic, elevated temperature, and malevolent acts, refer to Section 9B.2.6.

Design bases for the RB enveloping the containment structure and protecting it from external hazards and external beyond design basis scenarios are discussed in Section 9B.2.3.

9B.2.1.2 Structural Description of Containment

The BWRX-300 containment is a vertical cylinder comprised of the SCCV and a steel containment closure head (see Figure 9B-1). The containment structure is completely enclosed within the deeply embedded RB and includes personnel/equipment hatches, containment penetrations, and other safety components as shown in Figure 9B-1.

9B.2.1.2.1 Structural Description of Steel-Plate Composite Containment Vessel

As shown in Figure 9B-1, the SCCV structure consists of a cylindrical wall, mat foundation, and top slab constructed using DP-SC modules.

The SCCV DP-SC modules, including the inner and outer faceplates, diaphragm plates, steel headed stud anchors and concrete infill are part of the containment pressure boundary, with the modules inner faceplate also serving as a leak-tight liner.

The SCCV structure includes two airlocks (see Section 9B.2.1.2.2.2), penetrations (see Section 9B.2.1.2.2.3), and other safety components, such as the corium shield installed on the lower portion of the SCCV and RPV pedestal, also shown in Figure 9B-1. The corium shield protects the mat foundation and RPV pedestal from damage following core melt by preventing contact between the molten core and the DP-SC faceplates and concrete during severe accident conditions.

Besides the connection at the common mat foundation, the SCCV and the RB structures are also connected at the top slab and through the RB wing walls and floor slabs at intermediate floor levels as shown in Figure 9B-1. The SCCV top slab directly supports the reactor cavity pool and isolation condenser pools and extends beyond the containment wall boundary to the outer RB wall providing support for the fuel pool and cask pit.

A description of the steel containment closure head that covers the opening in the SCCV top slab over the RPV is provided in Section 9B.2.1.2.2.1.

9B.2.1.2.2 Class MC Steel Components of Containment

9B.2.1.2.2.1 Containment Closure Head

The containment closure head is a removable steel head anchored to the SCCV top slab using a pair of mating flanges. The containment closure head is part of the containment pressure boundary and forms part of the reactor cavity pool. The containment closure head is designed for removal during reactor refueling and for replacement prior to reactor operation.

Provisions are made to test the flange seals to ensure the containment closure head can perform its pressure retaining functions. Stainless steel cladding is fixed to the outer surface of the containment closure head to protect it from the water in the reactor cavity pool.

A structural overview of the containment closure head is shown in Figure 9B-1 and Figure 9B-4.

9B.2.1.2.2.2 Containment Airlocks

As shown in Figure 9B-1, two airlocks are provided in the SCCV structure to provide access to the upper and lower portions of the containment. These airlocks serve the dual purpose of allowing personnel access to the containment and for movement of equipment. Figure 9B-5 provides a structural overview of an airlock and its interface with the SCCV wall.

Each containment airlock has two pressure-seated doors interlocked to prevent simultaneous opening of both doors, and to ensure that one door is completely closed before the opposite door can be opened. The door operation is designed and constructed so either door may be operated from inside the containment, inside the lock, or from outside the containment. Also, each airlock has a barrel, which is an inner round portion that is slid into the fixed portion of the sleeve that is embedded in the SCCV wall. The barrel of the airlock can be removed if large equipment needs to be transported during outages.

9B.2.1.2.2.3 Penetrations

Figure 9B-1 depicts some of the containment penetrations, while Figure 9B-6 provides a structural overview of a typical penetration and its interface with the SCCV.

The major piping penetrations through the SCCV are associated with main steam and feedwater lines. Sleeve length for hot penetrations is designed to meet the concrete temperature limitations specified in NEDC-33926P, Section 6.5.1. SCCV electrical penetrations are sealed to the SCCV pressure boundary.

9B.2.1.3 Structural Analysis and Design Basis

9B.2.1.3.1 Applicable Codes, Standards and Other Specifications

The design, fabrication, construction, testing, and Inservice Inspection (ISI) of the SCCV and Class MC Steel Components of the containment conform to 10 CFR 50 and comply with the provisions of:

- GDC 1 - A design, manufacturing, and operating quality program is established for the containment structures commensurate with the importance of their safety functions as discussed in Section 9B.2.1.4.
- GDC 2 - The containment is designed to withstand the effects of a DBE, tornadoes, hurricanes, floods, tsunami, and seiches without loss of capability to perform safety functions as discussed in Section 9B.2.1.3.2.
- GDC 4 - The containment is designed to accommodate the effects of and be compatible with environmental conditions associated with normal operation, maintenance, testing, and postulated accidents, including Loss of Coolant Accidents (LOCAs) as discussed in Section 9B.2.1.3.2.
- GDC 16 - The SCCV, containment closure head and containment airlocks and penetrations are designed to provide a leak-tight barrier to prevent the uncontrolled release of radioactive effluents to the environment as discussed in Subsections 9B.2.1.3 and 9B.2.1.9.
- GDC 50 - The SCCV, containment closure head and containment airlocks and penetrations are designed to accommodate, without exceeding the design leakage rate and with sufficient margin, the calculated pressure and temperature conditions resulting from any LOCA as discussed in Subsections 9B.2.1.3 and 9B.2.1.9.
- GDC 51 - The SCCV, containment closure head and containment airlocks and penetrations are designed with sufficient margin to assure that under operating, maintenance, testing, and postulated accident conditions (1) the ferritic materials behave in a nonbrittle manner and (2) the probability of rapidly propagating fracture is minimized. The design reflects consideration of service temperatures and other conditions of the containment boundary material during operation, maintenance, testing, and postulated accident conditions, and the uncertainties in determining (1) material properties, (2) residual, steady state, and transient stresses, and (3) size of flaws as discussed in Subsections 9B.2.1.3 and 9B.2.1.9.
- GDC 52 - the SCCV, containment closure head and containment airlocks and penetrations are designed with provisions to conduct periodic integrated leakage rate testing at containment design pressure as discussed in Section 9B.2.1.9.4 and PSR Ch. 6, Section 6.5.10.
- GDC 53 - The containment is designed to permit (1) appropriate periodic inspection of all important areas, such as penetrations, (2) an appropriate surveillance program, and (3) periodic testing at containment design pressure of the leak tightness of penetrations as discussed in Section 9B.2.1.9.

The materials, analysis, design, fabrication, construction, examination and testing of the SCCV is in accordance with the provisions of NEDC-33926P, Section 6.0. These provisions are adapted from American Society of Mechanical Engineers (ASME) Boiler and Pressure Vessel Code (BPVC) 2021 Edition, Section III, “Rules for Construction of Nuclear Facility Components,” Division 2, “Rules for Construction of Nuclear Facility Components,” (Reference 9B-21), “Code for Concrete Containments,” Subsection CC, “Concrete Containments,” Articles CC-1000 through CC-6000, including Division 2 Appendices and changed based on analytical and engineering principles, including use of experimental results, to apply to a DP-SC containment vessel. Additional analysis and design requirements in NUREG-0800, SRP 3.8.1 “Concrete Containment,” (Reference 9B-22) and USNRC RG 1.136 for concrete containments are also met, as applicable.

The applicable sections of the remaining ASME BPVC, such as Section II, “Materials,” (Reference 9B-23), Section III, Subsection NCA, “General Requirements for Division 1 and Division 2,” (Reference 9B-24), Section V, “Non-Destructive Examination,” (Reference 9B-25), and Section IX, "Welding, Brazing, and Fusing Qualifications," (Reference 9B-26), are followed to the extent they apply to a DP-SC containment without reinforcing steel or tendons. ASME BPVC, Section XI, "Rules for Inservice Inspection of Nuclear Power Plant Components," Division 1 , "Rules for Inspection and Testing of Components of Light-Water-Cooled Plants," (Reference 9B-27), incorporated by reference in 10 CFR 50.55a(g)(4) 18 months before the date of issuance of the operating license for the initial 120-month ISI interval as described in 10 CFR 50.55a(g)(4)(i) is followed for the containment pre-service and periodic ISI and testing program.

The containment closure head, airlocks and penetrations are ASME Class MC components and are analyzed, designed, fabricated, constructed, inspected, and tested following the provisions of ASME BPVC Section III, Division 1, Subsection NE, "Class MC Components," (Reference 9B-28), Section XI, Subsection IWE, and 10 CFR 50 Appendix J in conformance with NUREG-0800, SRP 3.8.2, "Steel Containment," (Reference 9B-29).

Metal components backed by concrete are designed in accordance with NEDC-33926P, Section 6.0.

9B.2.1.3.1.1 Containment Code Jurisdictional Boundary

The BWRX-300 design codes jurisdictions are illustrated in Figure 9B-3.

As indicated in Section 6.1 of NEDC 33926P and in Figure 9B-3, the: SCCV boundary extends to the:

- Outside diameter of the SCCV wall from mat foundation to containment top slab including the welds connecting the SCCV with the RB structural members
- Portion of the mat foundation under SCCV including the welds connecting the SCCV portion of the mat foundation with the remaining part of the RB mat foundation
- Containment top slab from containment closure head opening to the outside diameter of the SCCV including the welds connecting the slab with the RB structural members

The code jurisdictional boundary for application of ASME BPVC Section III, Division 1, Subsection NE, Class MC to the containment closure head, airlocks and penetrations are shown in Figure 9B-4, Figure 9B-5 and Figure 9B-6, respectively.

The SCCV along with the containment closure head, airlocks, and penetrations, provide the primary containment function as a leak-tight pressure boundary confining radioactive substances in different plant conditions. Although the internal RPV support pedestal, bioshield and other containment internal structures are completely within the SCCV, these internal structures are non-pressure retaining structures and are, thus, designed per the requirements of ANSI/AISC N690.

The design of welds connecting the containment internal structures to the containment pressure boundary is per Section 6.11 of NEDC-33926P (and follows the quality assurance, welding procedures and inspection requirements of Sections 6.15 and 6.16 of NEDC-33926P. The connections of the RB walls and floors to the outside face of the SCCV wall are designed per Section 5.0 of NEDC-33926P, with the exception of attachment welds. Attachment welds connecting the RB walls and floors to the SCCV outside face are designed per Section 6.11 of NEDC-33926P and follow the quality assurance, welding procedures and inspection requirements of Sections 6.15 and 6.16 of NEDC-33926P.

9B.2.1.3.2 Loads and Load Combinations

9B.2.1.3.2.1 Containment Design Loads

The BWRX-300 containment is analyzed and designed for all credible conditions of loading, including normal loads, preoperational testing loads, loads during severe environmental conditions, loads during extreme environmental conditions, and loads during abnormal plant conditions. Load combinations considered in the design take into account the probability of occurrence, loading time history and sequence of loads, as applicable.

Loads used in the design of the BWRX-300 containment structures, comprised of the SCCV, containment closure head, and other Class MC components, satisfy the loading requirements of the applicable regulations, design codes and standards in Section 9B.2.1.3.1. These loads are in accordance with the provisions of ASME III Division 1, Subsection NE, and ASME III Division 2 and are consistent with the regulatory guidance of NUREG-0800, SRP 3.8.1 and SRP 3.8.2. Loads resulting from the application of prestress and from relief valve or other high energy device actuation are not considered in the design as these loads are not applicable to the BWRX-300 containment.

SCCV design loads include RPV reactions due to seismic loads, LOCA loads, and acoustic loads, as well as containment penetrations, brackets, and attachment loads. The connections between the containment, RB, and containment internal structures in the integrated FE model ensure the appropriate transfer of RPV reactions and other loads applied on the RB and containment internal structures to the SCCV structure.

Loads considered in the design of the BWRX-300 containment structures are:

- Normal Loads:
  - Dead load (D) which includes permanent dead weight of structural and shielding elements and permanently located equipment
  - Hydrostatic pressure of liquids in various pools (F)
  - Live loads (L, L_o) which include any moveable equipment loads and other loads that vary in intensity and occurrence
  - Indirect Snow (S) and Rain (R) loads
  - Thermal (T_o) effects and loads during normal operating, startup, or shutdown conditions
  - Pressure (P_o) loads resulting from the pressure difference between the interior and exterior of the containment, considering both interior pressure changes because of heating or cooling and exterior atmospheric pressure variations. External pressure loads resulting from pressure variation either inside or outside the containment (P_v), as defined in ASME BPVC Section III, Division 2, Subsection CC, Subparagraph CC-3221.1, are considered under the P_o load case.
  - Pipe reactions (R_o) during normal operating or shutdown conditions based on the most critical transient or steady-state conditions
  - Construction loads applied to the containment from start to completion of construction. The definitions for D, L and T_o given above are applicable, but are based on actual construction methods and/or conditions.
  - Indirect Lateral Soil and groundwater pressure loads (H)
- Pre-operational Testing Loads:
  - Thermal (T_t) effects and loads during the Structural Integrity Test (SIT) or Integrated Leak Rate Test (ILRT)
  - Test Pressure (P_t) loads applied during the SIT or ILRT
- Severe Environment Loads:
  - Indirect design Wind load (W) defined in PSR Ch. 3, Section 3.3.2

- Extreme Environmental Loads:
  - Indirect Tornado or Hurricane (W_t) loads defined in PSR Ch. 3, Section 3.3.2
  - DBE seismic (E_s) loads determined for the Standard Design conditions taking into account SSI effects, as discussed in PSR Ch. 3, Section 3.3.1, and including associated hydrodynamic loads and dynamic incremental soil pressures.
- Abnormal Plant Loads:
  - Accidental Thermal effects (T_a) due to LOCA
  - Accidental Pressure (P_a) loads within the containment generated by a LOCA
  - Accidental Pipe reaction loads (R_a) that consist of pipe reactions (including R_o) from thermal conditions generated by DBAs such as LOCA and DBE
  - Local effects on containment due to LOCA (R_r) and Blast Loads (R_b) which includes:
    - R_{rr} load on the containment generated by the reaction of a ruptured high-energy pipe during the postulated event of the DBA.
    - R_{rj} Load on the containment generated by the jet impingement from a ruptured high-energy pipe during the postulated event of the DBA.
    - R_{rm} load on the containment resulting from the impact of a ruptured high-energy pipe during the DBA.
    - Additional blast loads that may result from a postulated instantaneous break of a large pipe that could occur prior to the jet loads and that do need to be combined with the other loads.
  - Internal flooding loads resulting from a DBA (H_a).
  - Hard objects drop impact loadings, as applicable.

Loads associated with DEC representing a subset of BDBA conditions are discussed in Section 9B.2.6.

9B.2.1.3.2.2 Design Load Combinations for the SCCV

The SCCV is designed for the load combinations and load factors presented in Table 9B-2. Load combinations used in the design of the SCCV are in accordance with ASME BPVC, 2021 Edition, Section III, Division 2, Subsection CC, Subarticle CC-3230, supplemented by USNRC RG 1.136 and are consistent with the regulatory guidance of NUREG-0800, SRP 3.8.1.

9B.2.1.3.2.3 Design Load Combinations for the Containment Closure Head and Other Class MC Components

Load combinations and associated load factors used in the design of the containment Class MC components are in conformance with NUREG-0800, SRP 3.8.2 and USNRC RG 1.57, "Design Limits and Load Combinations for Metal Primary Reactor Containment System Components," (Reference 9B-30).

Table 9B-4 and Table 9B-5 summarize the load combinations and associated load factors used in the design of the containment closure head and other Class MC components, respectively.

As shown in Table 9B-4 and Table 9B-5, Class MC components combined loadings are categorized according to Level A, C and D service limits as defined in Subsection NE-3113 of ASME BPVC, Section III, Division 1. Because the Operating-Basis Earthquake (OBE) is set at one third of the DBE for design purposes in the Standard Design, OBE load combinations are not considered in the design. The OBE load combination involving post-flooding condition is considered for the design of Class MC components, excluding containment closure head, in accordance with NUREG-0800, SRP 3.8.2. OBE loads are also considered for cyclic loading considerations following the regulatory guidance of NUREG-0800, SRP 3.8.2.

Metal components backed by concrete are designed for the load combinations and associated load factors as summarized in Table 9B-2.

9B.2.1.3.3 Containment Design and Analysis Procedures

9B.2.1.3.3.1 Containment Structural Analysis Procedures

The BWRX-300 RB, containment, containment internal structures, and their common mat foundation are analyzed as one integrated structure. Analysis procedures for the integrated structure, including a description of the Three-Dimensional (3D) model and computer codes used in the analysis, and assumptions made on boundary conditions, are discussed in Section 9B.2.5.

As mentioned in Section 9B.2.1.2, the SCCV and RB structures are connected at several locations. The connections between the SCCV and the RB members in the integrated FE model are modeled to reflect the appropriate load transfer for vertical, lateral, and thermal loads. The SCCV connections are modeled and designed as discussed in NEDC-33926P, Sections 5.6 and 5.11, respectively.

Treatment of Axisymmetric and Non-axisymmetric Loads

ANSYS, the FE analysis software used to model and analyze the integrated RB, including the SCCV, evaluates the effects of the axisymmetric and non-axisymmetric loads applied on the SCCV 3D FE model by calculating the distribution of internal forces on the SCCV structural members.

The SCCV is shielded from the design wind, tornado, and hurricane by the RB, which completely encloses the structure. Forces from the design wind, tornado and hurricane are transmitted to the SCCV through the RB connections.

Indirect subgrade pressures are also transmitted to the SCCV through the RB connections. These pressures are obtained from the SSI analyses of the integrated RB discussed in PSR Ch. 3, Section 3.3.1.3. Methodology used to validate the subgrade pressure loads on the integrated RB is per Section 5.1.3 of NEDO-33914-A.

Major Penetrations

The SCCV major penetrations are described in Section 9B.2.1.2. These penetrations are included in the global integrated FE model used in the SSI analyses as stated in PSR Ch. 3, Section 3.3.1.3, and Section 9B.2.5. The state of stress and behavior of the SCCV around these openings is determined by use of acceptable analytical techniques, including but not limited to, the use of refined local FE models incorporating the imposed displacements calculated from the global integrated FE model discussed in Section 9B.2.5.

Variation in Physical Material Properties

The effects of variation in assumptions and material properties on the integrated RB SSI analysis results are considered as discussed in PSR Ch. 3, Section 3.3.1.3.

9B.2.1.3.3.2 Structural Design Method for SCCV

The SCCV structure design conforms to the requirements of NEDC-33926P, Section 6.0 and meets the acceptance criteria discussed in Section 9B.2.1.3.5. As stated in NEDC-33926P, the design guidelines for the SCCV are adapted from ASME BPVC Section III, Division 2 for concrete containments using current research data for steel-plate composite structures.

Membrane forces, shear forces and bending moments used in the design of the SCCV structure are obtained from the linear elastic SSI analyses for the integrated RB FE model discussed in Section 9B.2.5, for the loads and load combinations defined in Section 9B.2.1.3.2.

9B.2.1.3.3.3 Creep, Shrinkage, and Cracking of Concrete

The analysis of the SCCV considers the effects of creep, shrinkage, and cracking of concrete on the redistribution of forces and moments.

As stated in PSR Ch. 3, Section 3.3.1, stiffness calculations for structural modeling account for the expected state of stress and level of concrete cracking for different loading conditions during normal operation and accident conditions, including accident thermal loading.

The effective stiffnesses, geometric and material properties, for operational and accidental conditions, of SCCV elements are computed following Section N9.2.3 of ANSI/AISC N690 using the equations in Section 5.5 of NEDC-33926P.

9B.2.1.3.3.4 Tangential Shears/Stresses

Tangential (membrane) shears are captured in the 3D FE model of the SCCV. Tangential shear demands extracted from the FE model are assessed using the methodology in NEDC-33926P, Section 6.8.

9B.2.1.3.3.5 Corrosion Prevention

In accordance with NEDC-33926P, Section 5.15, the corrosion protection of the SCCV DP-SC modules is met by one or a combination of the following approaches to meet the design life and decommissioning time of the plant:

- Corrosion tolerance by adding a sacrificial thickness
- Protective paint system suitable to the surrounding environment
- Membrane coating system
- Impressed Current Cathodic Protection

The additional sacrificial thickness for corrosion tolerance is not considered in the DP-SC modules strength or stiffness estimates.

Guidelines in USNRC RG 1.54, “Service Level I, II, III, and In-Scope License Renewal Protection Coatings Applied to Nuclear Power Plants,” (Reference 9B-31) and ASTM D5144, “Standard Guide for Use of Protective Coating Standards in Nuclear Power Plants,” (Reference 9B-32), are used for application, maintenance, and qualification of permanent corrosion protective coatings, as applicable.

9B.2.1.3.4 Class MC Components Design and Analysis Procedures

Analysis and design procedures for the containment closure head, airlocks, and penetrations, are presented in Subsections 9B.2.1.3.4.1, 9B.2.1.3.4.2 and 9B.2.1.3.4.3, respectively.

The design of the Class MC steel components meets the acceptance criteria discussed in Section 9B.2.1.3.5, including buckling and fatigue evaluations as required.

The design by analysis utilizes demands obtained from the linear elastic SSI analyses for the integrated RB FE model discussed in Section 9B.2.5, for the loads and load combinations defined in Section 9B.2.1.3.2. For the treatment of non-axisymmetric and localized loads on Class MC components, refer to Section 9B.2.1.3.3.1.

Corrosion prevention methods considered for the containment closure head and Class MC components include the use of protective coatings and sacrificial thicknesses determined in accordance with Paragraph NE-3121 of ASME BPVC Section III, Division 1. The thickness of cladding fixed to the wetted surface of the containment closure head is determined in accordance with ASME BPVC Section III, Division 1, Paragraph NE 3122. The guidelines in USNRC RG 1.54 and ASTM D5144 are used as applicable for the application, maintenance, and qualification of corrosion protective coatings.

Effects of containment penetrations on the containment internal pressure capacity are considered as discussed in Section 9B.2.6.1.1 and in Section 6.23.1 of NEDC-33926P.

9B.2.1.3.4.1 Containment Closure Head

Design procedures for the containment closure head are shown in Figure 9B-7.

The containment closure head is analyzed using ANSYS and ACS SASSI computer programs and manual calculation. The stresses, including discontinuity stresses induced by the combination of external pressure or internal pressure, dead load, live load, thermal effects, and seismic loads, are evaluated. Minimum thickness evaluations, including local strengthening of the containment closure head at locations of attachments and openings, are performed in accordance with ASME BPVC, Section III, Division 1, Subsection NE, Subarticles NE-3100 and NE-3300. The required analyses and limits for the resulting stress intensities for the containment closure head are in accordance with ASME BPVC, Section III, Division 1, Subsection NE, Subarticle NE-3200.

Following the regulatory guidance of NUREG-0800, SRP 3.8.2, Section II.4, the evaluation of the containment pressure capacity considers buckling of the containment closure head under internal pressure as one of its limiting conditions. The compressive stress within the knuckle region caused by the internal pressure and the compression in other regions caused by other loads are limited to the allowable compressive stress values in accordance with Paragraph NE-3222 of ASME BPVC Section III, Division 1, or Code Case N-284-4, “Metal Containment Shell Buckling Design Methods, Class MC, TC, SC Construction Section III, Division 1 and Division 3,” (Reference 9B-33).

The containment closure head is also evaluated for fatigue, with the total number of stress cycles considered in the fatigue evaluation following the requirements of ASME BPVC, Section III, Division 1, Subsection NE, Subsubparagraph NE-3221.5(d). Because the containment closure head is submerged in the reactor cavity pool and can be exposed to water temperature above 50°C during accident, the fatigue evaluation of the containment closure head also follows the guidance of USNRC 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,” (Reference 9B-34).

9B.2.1.3.4.2 Containment Airlocks

Design procedures for the containment airlocks are shown in Figure 9B-8.

The containment airlocks are analyzed using ANSYS and/or manual calculation based on handbook formulas and tables. The discontinuity stresses induced by the combination of external, dead, and live loads, including the effects of earthquake loadings are evaluated. Minimum thickness evaluations for the containment airlocks are evaluated in accordance with ASME BPVC, Section III, Division 1, Subsection NE, Subarticles NE-3100 and NE-3300. The airlocks required analyses and limits for the resulting stress intensities are in accordance with Subarticle NE-3200 of ASME BPVC, Section III, Division 1, Subsection NE, including buckling evaluation per Code Case N-284-4 and fatigue evaluation as required.

The design and detailing of airlock components backed by concrete also meet the requirements of Section 6.12 of NEDC-33926P.

9B.2.1.3.4.3 Penetrations

Design procedures for the containment penetrations are shown in Figure 9B-8.

Penetrations are subjected to various combinations of piping reactions, mechanical, thermal, and seismic loads transmitted through the SCCV structure. The resulting forces due to various load combinations are combined with the effects of external and internal pressures. Penetrations are analyzed using ANSYS and/or manual calculation based on handbook formulas and tables for applicable loads and load combinations. Minimum thickness evaluations for the containment penetrations are evaluated in accordance with ASME BPVC, Section III, Division 1, Subsection NE, Subarticles NE-3100 and NE-3300. The penetrations required analysis and associated stress intensity limits are in accordance with Subarticle NE-3200 of ASME BPVC Section III, Division 1, Subsection NE, including buckling evaluation per Code Case N-284-4 and fatigue evaluation as required.

The design and detailing of penetration components backed by concrete also meet the requirements of Section 6.12 of NEDC-33926P.

9B.2.1.3.5 Structural Acceptance Criteria

9B.2.1.3.5.1 Design Basis Acceptance Criteria for SCCV

The acceptance criteria for the design of the SCCV are presented in Table 9B-6. Following the regulatory guidance of NUREG-0800, SRP 3.8.1, Subsection II.5 and as stated in NEDC-33926P, Section 6.6, these allowables are based on criteria in Tables CC-3432-1 and CC-3432-2 of ASME BPVC, Section III, Division 2, Subsection CC modified to include the allowable stresses for the steel component of DP-SC modules. The stress/strain acceptance criteria in Table 9B-6 allow the SCCV to remain elastic under service load combinations and below the range of general yield under factored loads.

9B.2.1.3.5.2 Design Basis Acceptance Criteria for Containment closure Head and Other Class MC Components

The acceptance criteria for the design of the Class MC components of the containment are the allowable stress limits specified in ASME BPVC, Section III, Division 1, Subsection NE-3220. Seismic design criteria for the containment Class MC components are discussed in Section 9B.2.1.3.5.3.

The structural acceptance criteria for the post-flooding condition, which is only applicable for Class MC components other than the containment closure head, is in accordance with NUREG-0800, SRP 3.8.2. Table 9B-7 and Table 9B-8 summarize the acceptance criteria for testing, design, Service Level A, C and D, and Post-flooding conditions, as applicable, for the containment closure head and other Class MC components, respectively.

Stability against compression buckling is in accordance with ASME BPVC, Section III, Division 1, Subparagraph NE-3222.1.

9B.2.1.3.5.3 Containment Seismic Design Criteria

The Seismic design criteria for the BWRX-300 containment, including the SCCV, containment closure head and other Class MC components are summarized in Table 3-1. The seismic design of the SCCV considers Limit State LS-D response in accordance with ASCE/SEI 43, ensuring an essentially elastic response when subjected to DBE.

9B.2.1.3.5.4 Containment Design Criteria for Impulsive and Impactive Loads

The SCCV is designed for impulsive and impactive loads per the regulatory guidelines of Appendix A of NUREG-0800, SRP 3.8.1. ASME BPVC, Section III, Division 2 provisions are not fully applicable to the DP-SC containment. Design allowable stresses and deformation limits developed for the SCCV design are provided in Section 5.8 of NEDC-33926P.

9B.2.1.3.5.5 Containment Robustness Acceptance Criteria

The Level Four D-in-D described in PSR Ch. 3, Section 3.1.7 requires that the containment design be robust to provide adequate protection for the confinement function, including the use of complementary design features to prevent accident progression and to mitigate the consequences of DECs and BDBAs. Refer to Section 9B.2.6.1 for a discussion of the robustness design and acceptance criteria for the BWRX-300 containment. These acceptance criteria are in accordance with USNRC SECY-93-087, "Policy, Technical and Licensing Issues Pertaining to Evolutionary and Advanced Light-water Reactor (ALWR) Designs," (Reference 9B-35), ensuring there is sufficient structural integrity to protect important systems in event of a design basis threat.

The leak tightness at the boundary of the containment structure, including the SCCV, containment closure head, and other Class MC components, under DEC internal pressure loads conforms to the regulatory guidance of USNRC RG 1.216, "Containment Structural Integrity Evaluation For Internal Pressure Loadings Above Design-Basis Pressure," (Reference 9B-36).

9B.2.1.3.6 Fire Protection

The containment structure is designed to have a 3-hour fire resistance rating. Fire protection features for the containment structure comply with requirements in RG 1.189 "Fire Protection for Nuclear Power Plants," (Reference 9B-37). Details associated with fire protection system design for this structure are provided in PSR Ch. 9A, Section 9A.6.

9B.2.1.4 Materials

The SCCV is constructed using DP-SC modules that are a type of steel-plate composite elements with diaphragm plates with holes attaching the two faceplates, headed studs steel anchors and concrete infill. Materials used in the construction of the SCCV DP-SC modules are in accordance with NEDC-33926P, Section 6.2 and USNRC RG 1.136.

To ensure the acceptability of the DP-SC modules for the construction of the SCCV, the NEDC- 33926P design requirements are supported by results from a test program performed under the National Reactor Innovation Center (NRIC) Advanced Construction Technology project in the United States. Mockups may be used to demonstrate the efficiency of fabrication, installation, and construction processes for use of DP-SC modules for nuclear applications.

The containment closure head and other Class MC components are made of steel. Stainless steel cladding is fixed to the outer surface of the containment closure head to protect it from the water in the reactor cavity pool above as discussed in Section 9B.2.1.2.

Steel materials used in the fabrication of the containment closure head and other Class MC components of the containment are in accordance with ASME Section III, Division 1, Subsection NE, Article NE-2000.

The impact testing requirements for SCCV steel faceplates are in accordance with NEDC-33926P. The impact testing requirements for containment closure head and Class MC components steel materials is in accordance with ASME Section III Subsection NE-2300 for Class MC components.

Design specifications covering materials are in sufficient detail to ensure the structural design requirements are met. Properties of materials used in the construction of the containment are provided in the following subsections.

9B.2.1.4.1 DP-SC Concrete Infill

Design compressive strength for the self-consolidating concrete used for the construction of the SCCV is determined as discussed in Section 5.2.1 of NEDC-33926P. Concrete is batched and placed in accordance with NEDC-33926P, Sections 6.2.1 and 6.15. Concrete and concrete constituents are examined and tested in accordance with NEDC-33926P, Section 6.16. Supplementary concrete sampling requirements are used as described in Section 3.2.2.1 of NEDO-33914-A to address the effect of the small volume of concrete placed for the BWRX-300.

Aggregates in concrete subjected to elevated temperatures for long periods of time do not contain pyretic materials.

9B.2.1.4.2 Containment DP-SC Steel

Containment DP-SC steel components conform to the following:

- Steel faceplates/diaphragm plates: ASME SA-738, Grade B or SA-537, Class 1 or Class 2, (ASME BPVC Section II, Part A, "Ferrous Material Specification," (Reference 9B-38)), or ASTM A709, "Standard Specification for Structural Steel for Bridges," (Reference 9B-39), Grade HPS 70W, per ASME Code Case N-763, "ASTM A 709-06, or Grade HPS 70W (HPS 485W) Plate Material Without Post-weld Heat Treatment as Containment Liner Material or Structural Attachments to the Containment Liner, Subsection CC," (Reference 9B-40).
- Studs: ASTM A108, "Standard Specification for Steel Bar, Carbon and Alloy, Cold Finished," (Reference 9B-41), as permitted by ASME BPVC Section III, Division 2, Subsection CC, MANDATORY APPENDIX D2-1, Table D2-I-2.2 and meeting the strength requirements of ASME BPVC Section III, Division 2, Subsection CC, Paragraph CC-2711, Table CC-2623.2-1.

9B.2.1.4.3 Structural Steel and Appurtenances for the SCCV

The structural steel and other appurtenances for the SCCV other than the primary DP-SC steel components covered above conform to the following:

- Structural steel embedded plate and attachments: ASTM A572, "Standard Specification for High-Strength Low-Alloy Columbium-Vanadium Structural Steel," (Reference 9B-42), Grade 50 per ASME Code Case N-632, "Use of ASTM A 572 Grades 50 and 65 for Structural Attachments to Class CC Containment Liners," (Reference 9B-43).
- Penetrations: ASME SA-516 (Reference 9B-38), Grade 60 or Grade 70, or SA-537 (Reference 9B-38), Class 2.

9B.2.1.4.4 Welding Materials

SCCV welding materials conform to the requirements of NEDC-33926P, Section 6.0 and ASME BPVC Section IX.

9B.2.1.4.5 Coatings

Corrosion protective coatings applied on the SCCV steel faceplates are discussed in Section 9B.2.1.3.3.5.

9B.2.1.4.6 Structural Steel and Appurtenances for the Containment Closure Head and Other Class MC Components

Structural steel and other appurtenances for the containment closure head and other Class MC components conform to the following:

- Structural Steel: ASME SA-516 Grade 60 or Grade 70, SA-537 Class 1 or Class 2 or SA-738 Grade B (Reference 9B-38)
- Non-pressure-retaining attachments: ASTM A572 Grade 50 as permitted by ASME BPVC, Section III, "Rules for Construction of Nuclear Facility Components," Division 1, Subsection NF, "Supports," (Reference 9B-44) referenced by ASME Section III, Division 1, Subsection NE
- Stainless Steel Cladding: ASME SA-264 steel-clad plate alloy and SA-240, Type 304L alloy for general application in accordance with ASME BPVC, Section II, Part A (Reference 9B-38)

- Bolting: ASME SA-193, Grade B7 or B8, or SA-437 Grade B4B (Reference 9B-38)

9B.2.1.4.7 Containment Quality Control

Quality control procedures are established for the containment structure in the construction, fabrication and installation specifications and implemented during fabrication, construction, installation, and inspection. These specifications cover the fabrication, furnishing, and installation of each structural item and specify the inspection and documentation requirements to ensure that the requirements of NEDC-33926P, NUREG-0800, SRP 3.8.2, Articles NE-4000 and NE-5000 of ASME Section III, Division 1, Subsection NE, USNRC RG 1.28, "Quality Assurance Program Criteria (Design and Construction)," (Reference 9B-45) and USNRC RG 1.136 are met.

9B.2.1.5 Interfaces with Other Equipment or Systems

The BWRX-300 containment is completely enclosed within the RB, with both structures resting on the common mat foundation that also supports the containment internal structures. In addition to sharing the common mat foundation, the containment is integrated with the RB structure at the top slab, at the connections of wing walls and at intermediate floor levels as shown in Figure 9B-1. In addition to interfacing with the RPV pedestal at the mat foundation, the containment also interfaces with the containment internal steel structures at various elevations (see Figure 9B-1) and with the main steam and feedwater piping and other electrical and piping lines penetrating through the containment wall. PSR Ch. 6, Section 6.5.6.3 provides more information on penetrations and piping into the containment structure.

The integrated RB model used in the SSI analyses discussed in Section 9B.2.5 appropriately incorporates the interfaces between the containment structure, containment internal structures, and RB to represent the vertical and lateral load transfer between the different structures for applicable internal and external loads.

The connections of the RB walls and floors to the outside face of the SCCV wall are designed as discussed in Section 9B.2.1.3.1.1.

9B.2.1.6 System and Equipment Operation

Refer to PSR Ch. 6, Section 6.5 for a discussion of containment systems designed to ensure its safety functions, including confinement, under accident conditions.

9B.2.1.7 High Level Construction Considerations

The SCCV is constructed using a modular construction technique as described in Section 9B.2.3.7.

Construction requirements are defined on drawings and/or specifications. Construction planning and constructability reviews are performed concurrently with the structural design.

Design development and construction method adopted for the deployment of the BWRX-300 plant takes into consideration CDM 2015 Regulations as described in PSR Ch. 14.

9B.2.1.8 Instrumentation and Control

In accordance with USNRC RG 1.12, seismic instrumentation is installed close to the top of containment as described in PSR Ch. 3, Section 3.3.1.5 to monitor the seismic motions during the lifecycle of the reactor facility.

Refer to PSR Ch. 6, Section 6.5, and PSR Ch. 7 for a discussion of systems and instrumentations used to monitor pressure inside the containment.

9B.2.1.9 Monitoring, Testing, Inspection and Maintenance

Containment safety functions are ensured throughout the life of the plant through monitoring, testing, inspections, and maintenance in compliance with 10 CFR 50.65 and USNRC RG 1.160, RG 1.136 and RG 1.216.

9B.2.1.9.1 Construction Testing and Inspection Requirements

The construction testing and examination requirements, including weld examination and qualification for the SCCV DP-SC modules are per Section 6.16 of NEDC-33926P supplemented by the concrete sampling requirements in NEDO-33914-A.

9B.2.1.9.2 Structural Integrity Test (SIT)/PRE-Operational Proof Test

The SCCV pre-service SIT plan and instrumentation are in compliance with Section 6.17 of NEDC-33926P and USNRC RG 1.136 and RG 1.216.

In accordance with NEDC-33926P, deformation, stress and strain measurements are made to evaluate the behavior of the containment and confirm that the actual structural response is within the limits predicted by analysis.

9B.2.1.9.3 Containment Pre-Service and Inservice Inspection

The SCCV pre-service and periodic in-service inspection plan is in accordance with Sections 6.17 and 6.22 of NEDC-33926P and meets the requirements of ASME BPVC, Section XI.

9B.2.1.9.4 Integrated Leak Rate Testing

The SCCV is designed such that the periodic Integrated Leak Rate Testing (ILRT) can be conducted at the design pressure as specified by GDC 52 of 10 CFR 50 Appendix A and discussed in PSR Ch. 6, Section 6.5.10.

9B.2.1.10 Radiological Aspects

The SCCV is designed to withstand the highest design basis accident pressure. The BWRX-300 SCCV is provided with overpressure protection by means of a containment inerting system as discussed in PSR Ch. 9A, Section 9A.4.2. The design basis accident pressure for containment is evaluated in PSR Ch. 15, Section 15.5 and the containment design pressure is discussed in PSR Ch. 6, Section 6.5.

Containment response to radiological events such as LOCA is discussed in PSR Ch. 15, Section 15.5.4.5.

As mentioned in PSR Ch. 6, Section 6.5, permanent radiation shielding is provided within and outside of the containment to limit doses to ALARP, with the SCCV structural concrete functioning as radiation shielding external to the containment, (see Section 9B.2.2.2 and Figure 9B-1). Radiation dose varies in the containment structure by locations and the SCCV minimum wall and top slab thickness and concrete infill mix are selected to provide adequate radiation shielding. For containment shielding evaluations, refer to PSR Ch. 12, Section 12.3.2. For other design features used for radiation protection, including the ALARP structures criteria observed in the BWRX-300 design, refer to PSR Ch. 12, Section 12.3.

The lower portion of the SCCV and RPV pedestal is also provided with a corium shield to prevent contact between the molten core and the DP-SC faceplates and concrete and RPV pedestal to mitigate the consequences of severe accident conditions as described in Section 9B.2.6.1.

Refer to PSR Ch. 6, Section 6.5 and PSR Ch. 7 for a discussion of engineered safety features, systems and instrumentation used to maintain the containment barrier for radiation control and protection.

9B.2.1.11 Performance and Safety Evaluation

The containment structure, including isolation of containment penetrations, is of a failsafe design such that confinement of radioactive material can be achieved for a minimum of seven (7) days under worst-case environmental conditions, without credit being taken for mitigating operator action.

To meet its functional and performance requirements listed in Section 9B.2.1.1, the containment structure is evaluated and designed for a comprehensive set of design loads described in Section 9B.2.1.3.2 and combinations (see Table 9B-2) using the method and design basis in Section 9B.2.1.3, with due consideration given of load type, probability of concurrence, loading time history and sequence of loads, as applicable.

To meet safety margin requirements in GDC 50 of 10 CFR 50 Appendix A, the containment design pressure is selected to bound DBAs with sufficient margin. As shown in PSR Ch. 15, Section 15.5.4.5, the containment peak pressure resulting from the most limiting design basis accident is at least 20% lower than the containment design pressure. Similarly, the containment design temperature bounds the accident peak values in PSR Ch. 15, Section 15.5.4.5.

The containment structure is enclosed within the RB, which protects the containment structure from the external hazards discussed in PSR Ch. 3, Section 3.3, including malevolent acts. Aircraft impact, with the BWRX-300 site as the target, due to malicious intent, terrorist acts or warfare are described in PSR Ch. 15, Section 15.8.3.5.5. As stated in PSR Ch. 15, Section 15.8.3.5.5 work is required to analyze the response of the plant to accidental aircraft impact events (see FAP Item PSR9B-279, Appendix B).

The containment structure is also designed for reaction forces stemming from the pipe movements, jet impact, pipe whip and other internal missiles described in PSR Ch. 3, Subsections 3.4.3 and 3.4.4.

The containment structure is designed as a floodable volume to assure core coverage in response to an accident, as described in PSR Ch. 6, Section 6.5.

Complying with GDC 3 of 10 CFR 50 Appendix A, containment walls are fire rated with steel plates exposed to fire as discussed in Section 6.20 of NEDC-33926P. Containment atmosphere is nitrogen-inerted during normal operation to preclude the initiation or propagation of a fire. Portable detection equipment and fire watches used inside containment during maintenance outages when the space is not inerted are discussed in PSR Ch. 9A, Section 9A.6.6.

The beyond design basis robustness to meet the DEC requirements of the containment structure are evaluated according to the criteria outlined in Section 9B.2.6.

9B.2.2 Containment Internal Structures

The BWRX-300 containment internal structures include the RPV pedestal, the bioshield surrounding the RPV pedestal, and the containment internal structural steel which consists of the CEPSS, and two support platforms as shown in Figure 9B-1.

9B.2.2.1 Structural Role and Safety Function

9B.2.2.1.1 Structural Role

The primary functions of the RPV pedestal are:

- To provide structural support to SSCs such as the RPV, RPV stabilizers, and miscellaneous platforms

- To provide radiation shielding to limit radiation dose within the applicable regulatory standards in different plant states, including normal operation, AOOs, DBAs, and DECs conditions

The primary functions of the bioshield are:

1. In conjunction with the RPV pedestal, to provide radiation shielding to limit radiation dose within the applicable regulatory standards in different plant states, including normal operation, AOOs, DBAs and DECs conditions
2. To provide structural support to miscellaneous platforms

The primary functions of the CEPSS are:

1. To provide supports for piping, RPV Stabilizers, RPV isolation valves including actuator, and other components
2. To serve as support for miscellaneous platforms and associated floor grating, ladders/stairs and other apparatus needed for personnel access and equipment inspection and maintenance

The primary functions of the support platforms are:

1. To provide support for components, including fans and coolers associated with the containment cooling system
2. To provide support for apparatus needed for equipment inspection and maintenance

9B.2.2.1.2 Safety Design Bases

Detailed SFRs for the Containment Internal Structures will be developed later when more detail is available for the Pre-Construction Safety Report.

The BWRX-300 containment internal structures are classified as SC1 consistent with the highest-class component they support and are BWRX-300 Seismic Category 1A structures as indicated in PSR Ch. 3, Table 3-1. The RPV pedestal and bioshield are also designed to provide radiation shielding to limit radiation dose during normal operation, AOOs, DBAs and DECs.

9B.2.2.2 Structural Description

The RPV pedestal is a cylindrical-shaped structure constructed using DP-SC modules that structurally supports the RPV and is integrated into the common mat foundation as shown in Figure 9B-1.

The RPV pedestal is equipped with a structural steel bracket that lines the top of the RPV pedestal where the RPV skirt is anchored using anchor bolts. The RPV pedestal also provides structural support for the RPV bottom stabilizers, the Control Rod Drive housing outer springs and the CEPSS. The part of the RPV pedestal surrounding the core region of the RPV helps to attenuate the radiation emanating from the RPV. Openings are provided in the RPV pedestal to permit the routing of necessary piping to the RPV, to permit ISI of the RPV and piping, and to ensure personnel access into the under-vessel region.

The bioshield is an independent cylindrical-shaped structure that surrounds the RPV pedestal (see Figure 9B-1) constructed using conventional steel-plate composite construction with ties and high-density concrete to reduce radiation shine. The bioshield is integrated with the common mat foundation and is separated from the RPV pedestal by a seismic gap. Similar to the RPV pedestal, openings are provided in the bioshield to permit the routing of necessary piping to the RPV, to permit in-service inspection of the RPV and piping, and to ensure personnel access into the under-vessel region.

The CEPSS is supported by the SCCV wall and the RPV pedestal as shown in Figure 9B-1. The CEPSS framing connects directly to the RPV pedestal and bypasses the bioshield so that the latter remains independent of the RPV pedestal. Shear lug support connections with the SCCV are used to allow free thermal expansion of the CEPSS, while resisting CEPSS lateral forces. The CEPSS consists of various structural components such as beams and columns as shown in Figure 9B-9.

9B.2.2.3 Structural Analysis and Design Basis

9B.2.2.3.1 Applicable Codes, Standards and Other Specifications

The design, fabrication, construction, testing, and ISI of the containment internal structures conform to 10 CFR 50 and comply with the provisions of GDCs 1, 2, 4, and 50.

The analysis, design, fabrication and testing of the DP-SC RPV pedestal is in accordance with ANSI/AISC N690, endorsed and modified per USNRC RG 1.243, including the modified design rules for the BWRX-300 non-containment steel-plate composite structures provided in Section 5.0 of NEDC-33926P (see Section 9B.2.3.3.1 for more details).

The analysis, design, fabrication and testing of the bioshield is in accordance with ANSI/AISC N690, including the supplemental requirements of USNRC RG 1.243 and USNRC RG 1.69, "Concrete Radiation Shields and Generic Shield Testing for Nuclear Power Plants," (Reference 9B-46).

The analysis, design, fabrication and testing of the of the containment internal structural steel is in accordance with ANSI/AISC N690.

Refer to **Error! Reference source not found.** for the jurisdictional boundary for the RPV pedestal, the bioshield and internal structural steel.

9B.2.2.3.2 Loads and Load Combinations

As the containment internal structures are completely contained within and are integrated with the RB and SCCV, the design of containment internal structures considers both design loads applied directly to the containment internal structures and those applied indirectly through the RB and SCCV.

9B.2.2.3.2.1 Design Loads

Refer to Subsections 9B.2.1.3.2 and 9B.2.3.3.2 for the description of design loads applicable for the SCCV and RB structures that are also generally applicable for the design of containment internal structures. As containment internal structures are inside the containment, some of the design loads applicable for the RB are not directly applicable for the containment internal structures (e.g., soil pressure, wind, tornado, and hurricane loads). Additionally, the internal flooding condition associated with post-accident flooding is not considered as noted in Table 9B-2).

The design loads also include the reactions from the RPV at the support locations on the containment internal structures and other bracket and attachment loads applicable during different plant conditions. The RPV lumped mass beam model representing the mass and stiffness properties of the RPV is included in the integrated FE model discussed in PSR Ch. 3, Section 3.3.1.3, and the dead load and seismic load reactions from the RPV are obtained directly from the static and seismic analyses. Other normal and accidental plant operating loads are applied to the model as reaction force loads.

9B.2.2.3.2.2 Design Load Combinations

Following the regulatory guidance of Subsection II.3 of NUREG-0800, SRP 3.8.3, "Concrete And Steel Internal Structures of Steel or Concrete Containments," (Reference 9B-47), load combinations and load factors for the design of the DP-SC structures and structural steel that form the containment internal structures are in accordance with ANSI/AISC N690, including the supplemental regulatory guidance of USNRC RG 1.243.

They are covered by the comprehensive set of load combinations listed in Table 9B-3 used for the design of the RB structure.

9B.2.2.3.3 Design and Analysis Procedures

9B.2.2.3.3.1 Structural Analysis Procedures

Analysis procedures for the containment internal structures are the same as those for the integrated RB structure discussed in Section 9B.2.5 as containment internal structures are included in the integrated FE model used in the analyses.

The RPV Pedestal in the integrated RB FE model includes major openings and penetrations in accordance with ANSI/AISC N690, Appendix N9. The state of stress and behavior of the RPV Pedestal around these openings is determined by use of analytical techniques.

The CEPSS is designed in accordance with ANSI/AISC N690 and is designed to carry piping dynamic loads without buckling and while remaining elastic. Pipe whip restraints attached to supporting CEPSS beams and columns are designed with the ability to sustain limited inelastic deformation without rupturing. SC1 SSCs affected by these inelastic deformations are evaluated to verify that their safety functions are not compromised. The support platforms shown in Figure 9B-1 are part of the integrated RB FE model and are analyzed and designed using the same analytical and design procedures as those for the CEPSS.

The connections between the containment internal steel structures and the RPV, RPV pedestal, bioshield and SCCV are appropriately modeled in the integrated FE model to reflect the appropriate load transfer for vertical, thermal, reactions forces, and lateral loads discussed in Section 9B.2.2.3.2.1.

9B.2.2.3.3.2 Structural Design Methods

For the design of containment internal structures, the design methodology is the same as that used for the design of the RB structure, discussed in Section 9B.2.3.3.3.

9B.2.2.3.4 Structural Acceptance Criteria

9B.2.2.3.4.1 Design Basis Acceptance Criteria

Acceptance criteria for the design of DP-SC RPV Pedestal, including welded and bolted connections, are in accordance with ANSI/AISC N690, Appendix N9, as endorsed by the regulatory guidance of USNRC RG 1.243, and the alternative design rules in Section 5.0 of NEDC-33926P.

The acceptance criteria for the bioshield and the containment internal structural steel are in accordance with ANSI/AISC N690.

9B.2.2.3.5 Fire Protection

The DP-SC RPV pedestal is designed to have a fire resistance rating not less than 3 hours. The fire rating requirement for the CEPSS and steel platforms is to be determined later and will be identified in the Pre-Construction Safety Report. Fire protection features for the containment internal structures comply with requirements in RG 1.189. Details associated with fire protection system design are provided in PSR Ch. 9A, Section 9A.6.

9B.2.2.4 Materials

The BWRX-300 RPV pedestal is a DP-SC structure, the bioshield is a conventional steel-plate composite structure, while the CEPSS and the support platforms are made of structural steel.

RPV pedestal DP-SC steel components conform to the following:

- Steel faceplates/diaphragm plate: ASTM A572 Grade 50, or ASTM A709 Grade HPS 70W
- Studs: ASTM A108 Type B
- Structural Steel embedded plate and attachments: ASTM A572 Grade 50

High density concrete used for shielding purposes is designed in accordance with USNRC RG 1.69.

CEPPS Structural steel conform to the following:

- Structural steel and connections: ASTM A572, A500, "Standard Specification for Cold-Formed Welded and Seamless Carbon Steel Structural Tubing in Rounds and Shapes," (Reference 9B-48) or A1085, "Standard Specification for Cold-Formed Welded Carbon Steel Hollow Structural Sections (HSS)," (Reference 9B-49)
- High strength structural steel plates: ASTM A572
- Structural Steel attachments to SCCV: ASTM A572 Grade 50 per ASME Code Case N-632.

Corrosion protection measures for BWRX-300 containment internal structures are the same as described in Section 9B.2.1.3.3.5.

The quality control requirements for the RPV pedestal are the same as those for the RB structure as described in Section 9B.2.3.4.8.

The quality control procedures for the CEPSS are in accordance with NUREG-0800, SRP 3.8.3, USNRC RG 1.28 and ANSI/AISC N690, Chapters NM and NN.

9B.2.2.5 Interfaces with Other Equipment or Systems

The RPV pedestal interfaces with the RPV through the RPV skirt support and horizontal stabilizers.

The CEPSS interfaces with the RPV at the upper stabilizer locations. Containment internal structures also interface with the support piping, equipment and general access platforms and stairs needed for personnel access and equipment inspection and maintenance.

9B.2.2.6 System and Equipment Operation

Supplemental equipment loads are added to the structural FE models to support mechanical and electrical equipment.

9B.2.2.7 High level Construction Considerations

The containment internal structures are constructed using a modular construction technique as described in Section 9B.2.3.7.

Construction requirements are defined on drawings and/or specifications. Construction planning and constructability reviews are performed concurrently with the structural design.

Design development and construction method adopted for the deployment of the BWRX-300 plant takes into consideration CDM 2015 Regulations as described in PSR Ch. 14.

9B.2.2.8 Instrumentation and Control

In accordance with USNRC RG 1.12, seismic instrumentation is installed on the containment internal structures as described in PSR Ch. 3, Section 3.3.1.5 to monitor the seismic motions during the lifecycle of the reactor facility.

For instrumentation used to monitor containment pressure and temperature, refer to PSR Ch. 6, Section 6.5.

9B.2.2.9 Monitoring, Testing, Inspection and Maintenance

A formal program of testing and ISI is not required for the containment internal structures as these structures are not directly related to the functioning of the containment system. However, during the operating life of the plant, the condition of the containment internal structures is monitored in compliance with 10 CFR 50.65 and in accordance with USNRC RG 1.160. To accommodate the ISI of containment internal structures, the design provides sufficient physical access to the structures.

The monitoring requirements for the RPV pedestal DP-SC modules are the same as those for the RB discussed in Section 5.18 of NEDC-33926P.

9B.2.2.10 Radiological Aspects

Penetrations through the containment wall are shielded to reduce radiation streaming in line with the ALARP criteria discussed in PSR Ch. 12, Section 12.1.1.

Corrosion protective coatings and systems are also applied on the exposed surface of the steel faceplates of the RPV pedestal to protect the metal from corrosion and facilitate decontamination.

9B.2.2.11 Performance and Safety Evaluation

To meet their functional and performance requirements listed in Section 9B.2.2.1, the containment internal structures are evaluated and designed for a comprehensive set of design loads described in Section 9B.2.2.3.2 and combinations (see Table 9B-3) using the method and design basis outlined in Section 9B.2.2.3, with due consideration given to the probability of concurrence, loading time history and sequence of loads, as applicable.

9B.2.3 Reactor Building Outside of Containment

9B.2.3.1 Structural Role and Safety Function

9B.2.3.1.1 Structural Role

The primary functions of the RB are:

1. To house the reactor vessel, the containment structure, the reactor support structure of the primary reactor system and fuel handling equipment, biological shielding, and associated equipment and structures.
2. To provide adequate space for the operation, maintenance, and removal of equipment housed within the containment structure during periodic maintenance.
3. To provide protection for SC1 SSCs from environmental and natural hazards phenomena, such as floods, winds, extreme winds (tornadoes and hurricanes), and earthquakes.
4. To provide protection for SC1 SSCs from external (e.g., explosions and missiles from nearly transportation/industry or aircraft impact) and internal (e.g., pipe break or heavy load drop) hazards.
5. To support habitability functions for the Secondary Control Room (SCR) such as radiological shielding and toxic gas isolation and passive cooling for human occupancy in addition to environmental qualification for equipment.

9B.2.3.1.2 Safety Design Bases

Detailed SFRs for the Reactor building outside of Containment will be developed later when more detail is available for the Pre-Construction Safety Report.

The BWRX-300 RB structure is classified as SC1 consistent with the highest-class component housed in the structure and is a BWRX-300 Seismic Category 1A structure as indicated in PSR Ch. 3, Table 3-1.

As mentioned in Section 9B.2.3.1.1, the main function of the RB structure is to protect the reactor vessel, the containment structure and SC1 SSCs from external hazards that might impair their safety functions during normal operation, AOOs, DBAs, and DECs in compliance with 10 CFR 50, Appendix A GDC 2.

External hazards considered in the design of the RB are described in PSR Ch. 3, Section 3.3.

9B.2.3.2 Structural Description

The RB is a deeply embedded cylindrical shaped building made of DP-SC floors and walls, with steel beams supporting the roof modules.

The below grade portion of the RB structure encloses the containment and protects the RPV, reactor support and SC1 SSCs, and the majority of vital and non-vital power supplies and equipment. While the majority of the RPV and SCCV is located below grade, the top elevation of the RPV and SCCV top slab is above grade. In addition, the above grade portion of RB structure houses the refueling floor, refueling and fuel handling systems, fuel pool, and polar crane.

The RB fuel pool provides for storage of new and spent fuel, along with the in-core components removed from the RPV during refueling. New fuel is staged in fuel storage racks located in the fuel pool. An area in the deep pit is used for loading of a spent fuel cask. The fuel pool racks are BWRX-300 Seismic Category 1A structures and are designed to withstand a DBE without failure of the basic structure or damage to the active region of irradiated fuel. For the performance characteristics, physical arrangement, operation, and design of the fuel storage racks, refer to PSR Ch. 9A, Section 9A.1.

The RB polar crane is composed of an overhead bridge of two deep girders supporting a trolley with a main and auxiliary hook. The top of circular rail is attached to the RB exterior wall and the bridge structure spans the full width of the refueling floor. The crane is classified as Seismic Category 2 to maintain structural integrity and is designed as a single failure proof system to eliminate the probability of a load drop event. For more information on the design of the RB polar crane, refer to PSR Ch. 9A, Section 9A.8.1. Figure 9B-1 presents an elevation sectional view of the structure while Figure 9B-3 depicts the separation of the RB structure code boundary from the containment pressure boundary discussed in Section 9B.2.1.3.1.1. RB structural boundary includes the following areas:

- DP-SC mat foundation between the SCCV wall and the RB exterior wall
- RB below grade (from the mat foundation) to the support for the polar crane, composed of DP-SC modules, including the roof which is supported by structural steel beams
- Isolation condenser pools, fuel pool, equipment pool, and the reactor cavity pool composed of DP-SC walls that are lined for corrosion protection with leakage detection system as applicable
- Rooms at several elevation levels outside the containment, structurally integral with the containment structure

As shown in Figure 9B-1, the SCCV and RB structures share a common foundation mat and are integrated at the connections of wing walls and elevated slabs between the mat foundation and the reactor cavity pool top slab elevation of the structure.

9B.2.3.3 Structural Analysis and Design Basis

9B.2.3.3.1 Applicable Codes, Standards and Other Specifications

To comply with 10 CFR 50 and the provisions of GDCs 1, 2 and 4, the materials, analysis, design, fabrication, construction, examination and testing of the RB structural steel and DP-SC components are in accordance with ANSI/AISC N690, including the supplemental requirements in USNRC RG 1.243, following the regulatory guidance of NUREG-0800, SRP 3.8.4, "Other Seismic Category I Structures," (Reference 9B-50).

The design of the RB DP-SC walls, slabs and mat foundation also meets the provisions of NEDC-33926P, Section 5.0. NEDC-33926P Section 5.0 provisions are adapted from ANSI/AISC N690 and adjusted to address the particularities of DP-SC construction. Sections 5.9 and 5.12 of NEDC-33926P justify why the provisions of ANSI/AISC N690 Appendix N9 can be extended to the design of the RB DP-SC floors and curved walls.

In addition to meeting the requirements of ANSI/AISC N690 and USNRC RG 1.243, the RB polar crane rail clips, runway girders, and support corbels are also designed to meet the requirements of ASCE/SEI 7, "Minimum Design Loads and Associated Criteria for Buildings and Other Structures," (Reference 9B-51), and Sections 4160 and 4460 of ASME NOG-1, "Cranes, Rules for Construction of Overhead and Gantry Cranes (Top Running Bridge, Multiple Girder)," (Reference 9B-52). Crane loading is developed in accordance with ASCE/SEI 7, Section 4.9.

9B.2.3.3.2 Loads and Load Combinations

In addition to the loads applicable directly to the RB, loads considered in the design of the RB include loads applied to the SCCV that are transferred to the RB structural components.

9B.2.3.3.2.1 Design Loads

The RB structure is analyzed and designed in accordance with ANSI/AISC N690, following the regulatory guidance of NUREG-0800, SRP 3.8.4.

Loads, such as accident pressure and thermal transient loads due to a LOCA, internal to SCCV are considered for the design of structural components of the RB that are integrated with the SCCV.

RB design loads consist of:

- Service category loads that occur during construction, pre-operational testing, or normal operation. They include:
  - Dead loads (D), which consist of the weight of structures, weight of permanently attached major equipment, tanks, machinery, and cranes; weight of piping, cable, cable trays and duct supports
  - Hydrostatic pressure (F) of liquids in various pools
  - Live loads (L, L_r), which consist of floor area loads, laydown loads, nuclear fuel, and equipment handling loads
  - Lateral Soil and groundwater pressure loads (H)
  - Snow/rain loads (S/R) discussed in PSR Ch. 3, Section 3.3.2
  - Normal plant operation and pre-operation pressure testing loads which consist of operation service pressure loads, pre-operation proof test pressure load, normal thermal conditions (T_o) and operation service pipe reaction loads (R_o)
  - Construction Loads
  - Settlement Loads

    - Crane Loads developed as discussed in Section 9B.2.3.3.1

- Abnormal and environmental category loads that occur during postulated accident and/or severe or extreme environmental events. They include:
    - Abnormal plant operation loads which include accident pressure (P_a) and thermal (T_a) loads, accident pipe reaction loads (R_a), missile generation, high energy pipe rupture loads (Y_r), pipe whip (Y_m), jet impingement from large pipe breaks (Y_j), blast pressure and compartment pressurization
    - Wind (W) and extreme wind (tornado loads or hurricane loads (W_t)) discussed in PSR Ch. 3, Section 3.3.2
    - Seismic loads (E_s) discussed in PSR Ch. 3, Section 3.3.1, including hydrodynamic loads on the pool walls calculated based on the approach described in ASCE/SEI 4, "Seismic Design of Safety-Related Nuclear Structures," (Reference 9B-53) and ACI 350.3, "Seismic Design of Liquid-Containing Concrete Structures and Commentary," (Reference 9B-54), and dynamic lateral pressures from the interaction of the deeply embedded structure with the surrounding subgrade.
- Hard objects drop impact loadings, as applicable
- Design Basis Threat loads discussed in Section 3.3.2, as applicable

Loads associated with DEC representing a subset of BDBA conditions are evaluated as discussed in Section 9B.2.6.

9B.2.3.3.2.2 Design Load Combinations

Load combinations and load factors for the design of the DP-SC structures and structural steel in the RB are in accordance with the provisions of ANSI/AISC N690, Chapter NB2.5 including the supplemental regulatory guidance of USNRC RG 1.243, Regulatory Positions 2.1 and 2.2.

Table 9B-3 provides the load combinations and load factors used in the design of the RB.

9B.2.3.3.3 Design and Analysis Procedures

9B.2.3.3.3.1 Structural Analysis Procedures

Refer to Section 9B.2.5 for analysis procedures.

9B.2.3.3.3.2 Structural Design Methods

The design of the RB DP-SC structures conforms to the requirements of NEDC-33926P, Section 5.0 and meets the acceptance criteria discussed in Section 9B.2.3.3.4. DP-SC connections in the RB are designed per the requirements in Section 5.11 of NEDC-33926P. The design of the RB steel structures is per ANSI/AISC N690 and meets the acceptance criteria discussed in Section 9B.2.3.3.4.

Membrane forces, shear forces and bending moments used in the design of the RB DP-SC and steel sections are obtained from the linear elastic SSI analyses for the integrated RB FE model discussed in Section 9B.2.5.

Results from the FE analyses are evaluated to identify critical cross-sections where maximum structural demands occur for different controlling loads and load combinations. Key responses reviewed include:

- Membrane forces for the SCCV
- In-plane shear demands at the base of major walls and at rock-soil interface elevation
- Vertical bending moments and out-of-plane shear demands on the RB outer shaft and SCCV walls, at base of walls and at intermediate floor elevations

- Out-of-plane demands for major floor slabs and RB foundation mat at mid-span and support locations

The structural demands at the critical locations are used to perform the design of the critical cross-sections and connections using the applicable codes of record.

9B.2.3.3.4 Structural Acceptance Criteria

9B.2.3.3.4.1 Design Basis Acceptance Criteria

The RB DP-SC structures and structural steel, including welded and bolted connections, are designed to meet the acceptance criteria outlined in ANSI/AISC N690 and in NEDC-33926P, Section 5.0.

The RB structure is evaluated for serviceability considerations including deflection, vibration, permanent deformation, concrete cracking, and settlement. Serviceability evaluations meet the acceptance criteria in ANSI/AISC N690, Chapter NL.

Seismic Design Criteria

The Seismic design criteria for the BWRX-300 RB are summarized in PSR Ch. 3, Table 3-1.

The seismic design of the RB structure considers Limit State LS-D response in accordance with ASCE/SEI 43, ensuring an essentially elastic response without any significant permanent deformations when subjected to the DBE and complying with ANSI/AISC N690.

The BWRX-300 RB structure meets the deformation acceptance criteria of ASCE/SEI 43, Section 5.2.3 and possesses ductility and energy absorbing capacity which permits inelastic deformations without failure under DECs.

Evaluation Criteria for Structure Interaction Under Seismic and Extreme Wind

The interaction of the RB structure with the adjacent RWB, CB, TB, Service Building and Reactor Auxiliary Structures is discussed in PSR Ch. 3, Subsections 3.3.1.3 and 3.3.2.4.

The stability of foundations under the DBE and extreme wind loads are checked following the criteria in Section 9B.1.1.3.2.

RB Design for Impulsive and Impactive Loads

The RB structure is designed for impulsive and impactive loads as discussed in Section 9B.2.3.3.2.1.

The RB is designed to withstand intrusion by regulatory defined threats by selecting design parameters for the DP-SC modules, such as steel thickness, steel ductility, and concrete thickness. The RB design for impulsive and impactive loads follows the provisions of ANSI/AISC N690 and the relevant regulatory guidance of USNRC RG 1.243. Allowable stresses and limits for the RB design for impulsive and impactive loads are provided in Section 5.8 of NEDC-33926P.

Criteria used to define the heavy loads considered in the RB design are described in PSR Ch. 3, Section 3.4.5.

9B.2.3.3.4.2 Robustness Acceptance Criteria for Reactor Building Structure

Refer to Section 9B.2.6.1.2 and 9B.2.6.2 for a detailed discussion of the robustness design and acceptance criteria for the BWRX-300 RB structure.

9B.2.3.3.5 Fire Protection

The RB structure is designed to have a 3-hour fire resistance rating. Fire protection features for the RB structure comply with requirements in RG 1.189. Details associated with fire protection system design for this structure are provided in PSR Ch. 9A, Section 9A.6.

9B.2.3.4 Materials

Materials used in the construction of the RB structure outside of containment are in accordance with Section 5.2 of NEDC-33926P and Section NA3 of ANSI/AISC N690.

The NRIC Demonstration Program Prototype test conclusions are used for demonstrating the structural performance of DP-SC modules and the adequacy of the proposed material requirements and design approaches discussed in Section 5.0 of NEDC-33926P.

9B.2.3.4.1 DP-SC Concrete Infill

Design compressive strength for the self-consolidating concrete used for the construction of the RB is determined as described in Section 5.2.1 of NEDC-33926P.

Concrete constituents used in the RB DP-SC modules are tested, batched, and mixed in accordance with ACI 349.

High density concrete used for shielding purposes is designed in accordance with USNRC RG 1.69. Aggregates used in high-density concrete for radiation shielding purposes conform to ASTM C637, “Standard Specification for Aggregates for Radiation-Shielding Concrete,” (Reference 9B-55), per ACI 349.

9B.2.3.4.2 DP-SC Steel

DP-SC steel components outside containment conform to the following:

- Steel faceplates/diaphragm plates: ASTM A572 Grade 50 or Grade 65
- Studs: ASTM A108, Type B

9B.2.3.4.3 Structural Steel

Structural steel conforms to ASTM A572 Grade 50 or Grade 65.

9B.2.3.4.4 Steel Connections

Bolts used in structural steel bolted connections are ASTM F3125, “Standard Specification for High Strength Structural Bolts and Assemblies, Steel and Alloy Steel, Heat Treated, Inch Dimension 120, Thousands of Pounds Per Square Inch (ksi) and 150 ksi Minimum Tensile Strength, and Metric Dimensions 830 MPA and 1040 MPA Minimum Tensile Strength,” (Reference 9B-56), Grade A325 Type 1.

Bolts of the following grades may be used otherwise, when deemed necessary:

- ASTM F3125, Grade A490 Type 1
- ASTM A354, “Standard Specification for Quenched and Tempered Alloy Steel Bolts, Studs, and Other Externally Threaded Fasteners,” (Reference 9B-57), Grade BD
- ASTM A449, “Standard Specification for Hex Cap Screws, Bolts and Studs, Steel, Heat Treated, 120/105/90 ksi Minimum Tensile Strength, General Use,” (Reference 9B-58), Type 1
- ASTM A193, “Standard Specification for Alloy-Steel and Stainless-Steel Bolting for High-Temperature or High-Pressure Service and Other Special Purpose Applications,” (Reference 9B-59), for high service temperature applications, grade is subject to specific design requirements.

9B.2.3.4.5 Welding

Welding materials for the RB structure are in compliance with the requirements of ANSI/AISC N690.

9B.2.3.4.6 Pool Liners and Appurtenances

Construction, fabrication, and installation of metallic liners, non-metallic liners, coatings, joint sealants, and water stops for structures where leak tightness is critical meet the regulatory guidance of NUREG-0800, SRP 9.1.2, “New and Spent Fuel Storage,” (Reference 9B-60).

Refer to Section 9B.2.4 for further discussion.

9B.2.3.4.7 Coatings and Other Corrosion Prevention

Corrosion protection measures used for the RB steel faceplates and structural steel are the same as those for the SCCV discussed in Section 9B.2.1.3.3.5.

9B.2.3.4.8 Quality Control

Quality control procedures are established and implemented during the construction and inspection phases of the RB structure. These procedures cover the fabrication, furnishing, and installation of each structural item in the RB and specify the inspection and documentation requirements in accordance with the requirements in ANSI/AISC N690, Section NA5 and Chapters NM and NN as required by USNRC RG 1.243 and discussed in Sections 5.16 and 5.17 of NEDC-33926P.

9B.2.3.5 Interfaces with Other Equipment or Systems

The RB is in close proximity to the RWB, CB, TB, Service Building and Reactor Auxiliary Structures. The interaction of the RB with the adjacent structures under seismic and extreme wind conditions is evaluated per criteria in PSR Ch. 3, Subsections 3.3.1.3 and 3.3.2.4 to ensure these structures will not collapse or collide with the RB and impair the safety functions of BWRX-300 Seismic Category 1A or 1B SSCs.

Adequate gaps are provided between the RB and the surrounding structures to prevent collision between structures. These gaps are evaluated along the entire height of the adjacent structures considering inelastic deformations, construction tolerances, and possible differential settlements as described in PSR Ch. 3, Section 3.3.1.3.

9B.2.3.6 System and Equipment Operation

Dynamic mass of the polar crane during the DBE is considered in the seismic analysis. Crane payload analysis is not included in this version of the PSR and will be included in the detailed design of the RB.

9B.2.3.7 High Level Construction Considerations

The BWRX-300 Seismic Category 1A structures are built using a modular construction technique using DP-SC (see Section 9B.2.5).

Construction requirements are defined on drawings and/or specifications. Construction planning and constructability reviews are performed concurrently with the structural design.

The quality control procedures used in the structural modularization process implemented in the construction of the DP-SC are outlined in Section 9B.2.3.4.8. These procedures are employed at the fabrication shop and the construction-site (both outside and inside the deep excavation pit necessary for the construction of RB), including pre-fabrication and pre-assembly, to ensure the DP-SC modular assemblies meet the necessary material quality, fabrication, and installation requirements per the applicable code of record.

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

For plant construction and commissioning activities, refer to PSR Ch. 14.

Design development and construction method adopted for the deployment of the BWRX-300 plant takes into consideration CDM 2015 Regulations as described in PSR Ch. 14.

9B.2.3.8 Instrumentation and Control

Not applicable to design of RB structure outside of the containment.

9B.2.3.9 Monitoring, Testing, Inspection and Maintenance

Periodic inspection, and in-service monitoring programs are implemented to ensure the RB structure continues to meet its functional and performance requirements in compliance with 10 CFR 50.65 and following the regulatory guidance of USNRC RG 1.160.

The design of the RB provides sufficient physical access to accommodate ISI of the structure. Non-destructive testing and remote visual monitoring are used to monitor conditions in inaccessible and high radiation areas.

The aging management, ISI and testing of the RB DP-SC structures, in accessible and inaccessible areas, are performed as discussed in NEDC-33926P, Section 5.18. Inspections and monitoring during construction and commissioning are performed as described in Section 3.2 of NEDO-33914-A. The post-construction testing and in-service surveillance programs for the RB below grade wall and foundation include monitoring of groundwater chemistry, settlements and differential displacements as discussed in Sections 3.3 and 3.4 of NEDO-33914-A.

Field monitoring of the settlement and tilt of the RB shaft is performed as described in Section 9B.1.1.9.

9B.2.3.10 Radiological Aspects

Design measures that enable the RB to provide flow resistance in contaminated areas that hold up radioactive releases from the containment are discussed in PSR Ch. 12, Section 12.3. They include the use of suitable shielding for the fuel pool and associated equipment in addition to using a system for detecting fuel pool leakage.

The design of the SCCV, RPV pedestal and the bioshield, including their selected materials and geometric configuration, ensures that these structures provide adequate shielding from radiation emanating from the fuel core in the RPV.

Shielding of personnel in the SCR is discussed in PSR Ch. 12, Section 12.3.2.

9B.2.3.11 Performance and Safety Evaluation

To meet its functional and performance requirements listed in Section 9B.2.3.1, the RB structure is evaluated and designed for a comprehensive set of design loads described in Section 9B.2.3.3.2 and combinations (see Table 9B-3) using the method and design basis described in Section 9B.2.3.3, with due consideration given of the probability of concurrence, loading time history and sequence of loads, as applicable.

To prevent and resist the spread of internal fires within the building, the RB structure is supplied with fire detection and suppression systems. For more details on the fire protection system design refer to PSR Ch. 9A, Section 9A.6.

RB walls, floors, ceilings, doors, and penetrations are fire rated, as required under the fire hazard assessment. The fire rating requirements and DP-SC modules capacity under fire are as described in Section 5.13 of NEDC-33926P. Fire barriers are also employed to prevent the spreading of fires from a non-safety room to a room with safety class system or equipment and between redundant safety class fire zones.

For flood protection measures for the RB, refer to PSR Ch. 3, Subsections 3.3.2.6 and 3.4.2.

The circular design of the embedded section of the BWRX-300 RB structure provides significant torsional support for the building. Accidental torsional moment demands at each floor level are computed as described in PSR Ch. 3, Section 3.3.1.3.

As mentioned in Section 9B.1.1.3.2, due to its deep embedment, the sliding and overturning stability of the integrated RB is considered satisfied without the need for explicit sliding and overturning stability evaluations.

The RB is designed to resist global failure, perforation, spalling, and fuel intrusion from the regulatory defined threats to protect the containment structure housed within its structure. For more details on the robustness of the RB structure against malevolent acts, refer to the PSR Security Annex (PSR Ch. 25). Aircraft impact, with the BWRX-300 site as the target, due to malicious intent, terrorist acts or warfare are described in PSR Ch. 15, Section 15.8.3.5.5. As stated in PSR Ch. 15, Section 15.8.3.5.5, work is required to analyze the response of the plant to accidental aircraft impact events (see FAP Item PSR9B-279, Appendix B).

The beyond design basis robustness to meet the DEC requirements of the RB structure are evaluated according to the criteria outlined in Section 9B.2.6.

9B.2.4 Reactor Building Pools and Liners

The BWRX-300 RB pools are classified as SC1 consistent with the highest-class component housed in the pools.

The BWRX-300 RB has a number of pools located in the upper part of the building including:

1. Fuel pool
2. Reactor cavity pool
3. Equipment pool
4. Isolation condenser pools

In addition, there are two skimmer tanks embedded into the pool walls between the Fuel Pool and Reactor Cavity.

9B.2.4.1 Structural Role and Safety Function

9B.2.4.1.1 Safety Functional Requirements

The structural role of the pool walls, base slab and liners is to provide structural support to the water mass and pool inventories, and to ensure leak-tightness of pools.

The pools and liners perform an important safety role, and directly contribute to the fulfilment of Fundamental Safety Functions. Detailed SFRs for the pools and liners will be developed later when more detail is available for the Pre-Construction Safety Report.

Pool liners are selected to withstand the environmental conditions they are to encounter during the life of the facility. Pool liners, and their connections, are designed to sustain high temperatures, resist radiation exposure, resist impactive and abrasive loads, maintain leak tightness and maintain the pool inventory at acceptable levels during normal operation and after design basis events. Pool liner materials or coatings are also designed to resist flaking or chipping to avoid foreign materials from being transported to the reactor core and clogging the Passive Containment Cooling System.

9B.2.4.2 Structural Description

Section is to be completed later when more detail is available for the Pre-Construction Safety Report.

9B.2.4.3 Structural Analysis and Design Basis

Section is to be completed later when more detail is available for the Pre-Construction Safety Report.

9B.2.4.4 Materials

Liner plates may be of the following type and grade:

- Carbon steel with stainless clad: ASTM A264, "Standard Specification for Stainless Chromium-Nickel Steel-Clad Plate," (Reference 9B-61) (A516, "Standard Specification for Pressure Vessel Plates, Carbon Steel, for Moderate and Lower-Temperature Service," (Reference 9B-62), Grade 70 plus A240, "Standard Specification for Chromium and Chromium-Nickel Stainless Steel Plate, Sheet, and Strip for Pressure Vessels and for General Applications," ASTM International, (Reference 9B-63), Type 304L)
- Stainless steel: ASTM A240 Type 304L.

9B.2.4.5 Interfaces with Other Equipment or Systems

Section is to be completed later when more detail is available for the Pre-Construction Safety Report.

9B.2.4.6 System and Equipment Operation

Section is not applicable to pools and liners.

9B.2.4.7 High Level Construction Considerations

Construction, fabrication, and installation of metallic liners, non-metallic liners, coatings, joint sealants, and water stops for structures where leak tightness is critical meet the regulatory guidance of NUREG-0800, SRP 9.1.2.

Liner plates, if used as forms, shall be designed for the lateral pressure corresponding to the rate of concrete placement or shall be provided with a suitable bracing system as noted on the drawings. Temporary loads on liners are not additive to normal design loads.

Construction requirements are defined on drawings and/or specifications. Construction planning and constructability reviews are performed concurrently with the structural design.

Design development and construction method adopted for the deployment of the BWRX-300 plant takes into consideration CDM 2015 Regulations as described in PSR Ch. 14.

9B.2.4.8 Instrumentation and Control

Section is to be completed later when more detail is available for the Pre-Construction Safety Report.

9B.2.4.9 Monitoring, Testing, Inspection and Maintenance

Section is to be completed later when more detail is available for the Pre-Construction Safety Report.

9B.2.4.10 Radiological Aspects

Section is to be completed later when more detail is available for the Pre-Construction Safety Report.

9B.2.4.11 Performance and Safety Evaluation

Section is to be completed later reference when more detail is available for the Pre-Construction Safety Report.

9B.2.5 Structural Analysis Criteria for BWRX-300 Seismic Category 1A Structures

The RB, containment and the containment internal structures are analyzed as one integrated structure, using ANSYS and ACS SASSI computer programs, to determine structural design demands resulting from various design loads and design load combinations. Evidence of qualification of these computer programs, including a description of the programs and extent of use, is presented in PSR Ch. 3, Appendix D.

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

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

These analyses are performed on integrated RB structural models that have the same node and element type configurations. The use of linear elastic models with identical FE configuration enables the demands obtained from different analysis cases to be combined on an element-by-element basis for the applicable design load combinations per governing design codes.

Static analyses provide design demands on the RB integrated structures from dead loads, live loads, earth pressure loads, hydrostatic and hydrodynamic loads, severe and extreme environmental loads, plant operating loads during normal operation, testing and abnormal plant conditions. Thermal analyses provide stress demands due to normal operating and accidental load conditions. DBE seismic demands are obtained directly from the results of one-step approach SSI seismic analyses discussed in PSR Ch. 3, Section 3.3.1.3.

The effect of interaction with the surrounding subgrade is incorporated in the analyses of the deeply embedded integrated RB by considering the surrounding soil and rock as a layered half-space continuum. The geotechnical design parameters used as input for the static and thermal analyses are developed as described in Section 9B.1.1.3.2.

9B.2.5.1 FE Model of Integrated RB Structure

To determine internal forces resulting from various loads and loading combinations, a detailed structural model is developed for the integrated RB, containment, and containment internal structures, including their foundations, penetrations, and openings, following the general FE modeling guidelines discussed in PSR Ch. 3, Section 3.3.1.3 and NEDO-33914-A, Section 5.1.1. The model adequately represents the RB structural configuration for all main structural members and meets the mesh refinement and quality attributes required for calculation of structural stress demands.

Materials properties assigned to the integrated RB model depend on the analyzed loads and resulting stress responses. Unit weight properties are assigned to the models used for the 1-g static SSI analyses to adequately simulate gravity and earth pressure loads. The dynamic model of the integrated RB used for the seismic SSI analyses is assigned seismic mass inertia properties as discussed in PSR Ch. 3, Section 3.3.1.3.

As discussed in PSR Ch. 3, Section 3.3.1.3, stiffness properties are assigned to the SCCV and RB to reflect effective stiffness for load combinations without accidental thermal load. For load combinations with accidental thermal load, reduced stiffness is considered to account for the concrete cracking effects on the redistribution of forces and moments. Spring elements are also used in the integrated FE element model to represent the stiffness of the connections between the different structural members that are designed to relieve stresses due to thermal expansion.

9B.2.5.2 1-g Static Soil-Structure Interaction Analyses

Stress demands for the design of the integrated RB structure from dead loads and earth pressure design loads are obtained by applying the Earth gravity (1-g) load in the vertical direction to the SSI model described in Section 9B.2.5. The 1-g static SSI analyses utilize the same sub-structuring method as the seismic SSI analyses described in PSR Ch. 3, Section 3.3.1.3. LB equivalent linear stiffness properties and UB unit weight properties assigned to the subgrade model used in the analyses are discussed in Section 9B.1.1.3.2.

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

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

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

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

Contact springs are used at the interfaces of the RB structure with the surrounding subgrade as discussed in PSR Ch. 3, Section 3.3.1.3. In accordance with the FE modeling guidance in NEDO-33914-A, Section 5.1.1, the following stiffness properties are assigned to the contact springs in the models used for the 1-g static SSI analyses to provide UB lateral soil pressures on the RB below grade exterior wall:

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

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

9B.2.5.2.1 Subgrade Modeling Assumptions for Deeply Embedded Reactor Building

Per NEDO-33914-A, Section 5.1.2, the following assumptions related to the modeling of the subgrade are introduced in the 1-g static SSI analyses to enable an efficient calculation of stress demands on the RB structure due to pressure loads from soil and rock surrounding and supporting the RB shaft:

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

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

The effects of non-linearities at soil-structure interfaces are addressed by using elastic contact spring stiffness properties that provide bounding structural demands following the guidelines of Section 5.1.2 of NEDO-33914-A.

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

9B.2.5.2.2 RB Design Earth Pressure Load Validation

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

In accordance with requirements in NEDO-33914-A, Section 4, FIAs are performed on models representative of the non-linear constitutive behavior of soil and rock materials surrounding the RB shaft. The models employ non-linear interface modeling features capable of capturing the effects of non-linearities at the subgrade structure contact surfaces. The results of the FIAs are used for validation of the design earth pressures following the guidance of Section 5.1.3 of NEDO-33914-A.

9B.2.5.3 Static and Quasi-Static Load Analyses

The static and quasi-static analyses provide stress demands due to:

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

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

- Super-element representing LB stiffness of the subgrade surrounding the RB, and
- Integrated FE model of the RB, containment and containment internal structures described in Section 9B.2.5.1.

The super-elements define the stiffness of the subgrade at the nodes of the RB interfaces with the surrounding soil. The stiffness properties of the super-elements are developed using a layered 3-D solid FE model. Subgrade stiffness properties assigned to the super-elements are described in Section 9B.1.1.3.2. To adequately simulate half-space boundary conditions, the depth of these models is deeper than three times the largest foundation dimension, or two times the depth of the RB embedment. The horizontal extent of these models is more than three times the RB shaft diameter.

The nodes of the super-element are coincident with the nodes of the integrated RB FE structural model. The coincident super-element and structural model nodes are connected by contact spring elements as described in Section 9B.2.5.2. LB stiffness properties are assigned to these contact spring elements to yield larger structural deformations and conservative design stress demands. Equivalent linear subgrade stiffness properties assigned for the subgrade stiffness impedance static analyses are discussed in Section 9B.1.1.3.2.

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

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

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

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

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

9B.2.5.4 Thermal Stress Analyses

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

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

For the thermal analyses, UB stiffness properties are assigned to the super-element modeling the subgrade and to the contact elements modeling the soil-structure interfaces resulting in conservative thermal stress demands for the design of the RB and containment structures. Equivalent linear subgrade stiffness properties assigned for the thermal stress analyses are discussed in Section 9B.1.1.3.2.

9B.2.6 Beyond Design Basis Considerations for BWRX-300 Seismic Category 1A Structures

Consistent with the Level Four D-in-D requirements discussed in PSR Ch. 3, Section 3.1.7, the BWRX-300 containment, and RB are robust structures, tolerant of a large spectrum of faults with a gradual degradation in their effectiveness and would not fail catastrophically under operational states, DBAs and DECs.

Evaluations performed to establish an understanding of safety margins, or the robustness of the design are consistent with the regulatory guidance of NUREG-0800, SRP 19.0, “Probabilistic Risk Assessment and Severe Accident Evaluation for New Reactors,” (Reference 9B-64).

9B.2.6.1 Design Extension Conditions

Deterministic safety analyses are used to determine the applicable DECs and evaluate the consequences. A Best Estimate (BE) approach is used to obtain a reasonable confidence in the assessed response to DECs.

A reasonable level of survivability of the structure under postulated DECs is demonstrated. Less stringent assumptions than those applied for design basis, such as the permissible variances, may be used when evaluating SSC performance under DECs.

9B.2.6.1.1 Containment Severe Accident Design Extension Condition Evaluations

Complying with GDC 51 of 10 CFR 50 Appendix A, the containment structure is designed to possess ductility and energy absorbing capacity, which permits inelastic deformation without failure under DECs.

The beyond design basis evaluations of the containment follow the guidance of NUREG-0800, SRP 19 to ensure the structural integrity and leak tightness of the containment structure under all applicable DEC loading cases.

9B.2.6.1.1.1 Containment Ultimate Pressure Capacity

The ultimate pressure capacity of the containment structure, including the SCCV, containment closure head and penetrations, is determined to ensure its structural integrity and leak tightness under beyond design basis internal pressure loads following the regulatory guidance of NUREG 0800, SRP 3.8.1 and USNRC RG 1.216.

The containment ultimate pressure capacity is obtained from the results of static FE analysis performed as described in NEDC-33926P, Section 6.23.1, consistent with the guidelines of Regulatory Position 1 of USNRC RG 1.216.

9B.2.6.1.1.2 Robustness Against Combustible Gas Pressure Loads

As stated in NEDC-33922P-A, “BWRX-300 Containment Evaluation Method,” (Reference 9B-65) and in PSR Ch. 6, Sections 6.5 and PSR Ch. 15, Section 15.5.4.5, the containment design prevents combustible gases from creating a concern for deflagration in the containment during a design basis accident. The containment structural robustness against beyond design basis combustible gas pressure loads is evaluated using the methodology discussed in NEDC-33926P, Section 6.23.2.

9B.2.6.1.1.3 Containment Severe Accident Performance Goal

Per USNRC SECY-93-087, the BWRX-300 containment under the more likely severe accident conditions:

- Maintains its role as a reliable leak-tight barrier for a minimum of 24 hours following the onset of core damage, and
- Continues to provide a barrier against the uncontrolled release of fission products following the initial 24-hour period

The methodology used to evaluate the robustness of the containment is per Regulatory Position 3 of USNRC RG 1.216 and as described in NEDC-33926P, Section 6.23.3. During an extremely improbable severe accident in the BWRX-300, molten core debris may be present on the containment floor. A protective layer of refractory concrete (as shown in Figure 9B-1) prevents corium from degrading the SCCV inner steel faceplate that acts as the primary leak-tight boundary. Additional protection is provided by the outer steel faceplate for the SCCV foundation mat. The lower portion of the SCCV structure design has a provision for the installation of a severe accident core melt capture and retention structure with a spreadable area to prevent contact between the molten core and the containment liner and concrete.

9B.2.6.1.2 Beyond Design Basis Seismic Robustness

The design of the BWRX-300 Seismic Category 1A and 1B SSCs credited to function during and after a Beyond-Design Basis Earthquake (BDBE) ensures their capability to maintain their structural integrity and to perform their intended safety function.

To demonstrate compliance with the regulatory guidance of NUREG-0800, SRP 19.0, a seismic probabilistic safety assessment is performed to demonstrate an inherent capability of safe shutdown in response to BDBE with a plant level of High Confidence of Low Probability of Failure (HCLPF) of at least 1.67 times the peak ground acceleration of a DBE.

The methodology in Electrical Power Research Institute (EPRI) TR-103959, “Methodology for Developing Seismic Fragilities,” (Reference 9B-66) and TR-3002012994, “Seismic Fragility and Seismic Margin Guidance for Seismic Probabilistic Risk Assessments,” (Reference 9B-67) is used for the evaluations of seismic fragilities of BWRX-300 Seismic Category 1A and 1B SSCs.

To ensure adequate margins for consideration of robustness against BDBE, the seismic design satisfies the ductility detailing and design requirements for steel and steel-plate composite structures of ANSI/AISC N690, with the supplementary guidance of USNRC RG 1.243 and NEDC-33926P. Reinforced concrete ductile detailing is in line with ACI 349, PSR Ch. 21.

9B.2.6.2 Design for Malevolent Acts

The BWRX-300 uses a security by design process that involves security reviews during plant design to resolve Design Basis Threat (DBT) and Beyond Design Basis Threat (BDBT) security issues at the earliest stage, when changes have the least effect on cost and performance. Placement and number of doors, wall thicknesses to optimize resistance to explosive breaching, and equipment placement to facilitate better target set diversity are all achievable when security is integrated at an early stage. Continual design reviews against the DBT and BDBT capabilities during the entire design evolution ensure that emergent issues are identified and addressed as early in the process as possible.

The defensive strategy approach focuses on protecting the passive plant features and other key reactor components from hostile action by creating a robust perimeter. By analyzing the potential adversary pathways to critical components, determining adversary resources required to execute the path, and slowing the adversary movements and depleting the adversaries' resources before the path can be completed to the extent possible, the design limits the ability of malicious individuals to cause damage to key systems. This, along with the inherent slower accident progression of the BWRX-300 reactor, reduces or eliminates the reliance on immediate on-site armed responders to prevent substantial off-site radiological releases, which allows for longer term off-site response, interdiction, and neutralization.

9B.2.6.2.1 Malevolent Acts Design Methods

The BWRX-300 design applies the Nuclear Energy Institute's methodology in NEI 07-13, "Methodology for Performing Aircraft Impact Assessments for New Plant Designs," (Reference 9B-68) for aircraft crash evaluations with UK input and other detailed computer analytical methods, where appropriate, to evaluate the consequences of regulatory defined threats on a BWRX-300 reactor site. The UK acceptance criteria are then applied to the results.

The design considers two types of structural failure modes with distinct loading characteristics and structural responses:

1. Local effects that in general would not result in structural collapse but may affect the functions of SC1 SSCs
2. Global failure modes characterized by major structural damage, such as significant perforation or collapse of large portions of the building walls, floors, and load carrying frames

These failure modes are considered separately with consideration given that for some threats, such as an aircraft crash, they may act simultaneously or quasi-simultaneously.

9B.2.6.2.2 Malevolent Acts Design Acceptance Criteria

The BWRX-300 design ensures the availability of the safety functions and capabilities of SSCs which failure can result in significant radioactive releases during and after a DBT event. The design provides for the ongoing availability of fundamental safety functions of these SSCs during BDBTs depending on the severity of the threat.

For more severe BDBTs, the design ensures a sufficient structural integrity to protect important safety systems that provide safe shutdown path. At least one means of reactor shutdown and core cooling are provided in the event of extreme BDBT.

The acceptance criteria for local and global structural response are satisfied simultaneously.

Design criteria for the BWRX-300 RB specifies no global failure, no perforation, no spalling, and no fuel intrusion from the regulatory defined threats.

The PSR Security Annex provides additional discussion on protection for malevolent acts.

9B.3 Other Structures

It should be noted that the level of maturity of the other structures is lower than that of the integrated RB.

In addition, although a Standard Design exists for these structures which has been based on local United States (U.S.) non-nuclear building codes (with the exception of the RWB which is designed in line with a specific USNRC regulatory Guide, as discussed in Section 9B.3.1.3.2), the assumption is that local U.S. building codes are not applicable in the UK and therefore UK specific building codes are referred to in this section. The applicability of the Standard Design for these buildings in the UK will be justified through detailed analysis and assessment at Site Specific stage.

9B.3.1 Radwaste Building

The RWB houses a portion of the Offgas 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 location is shown in PSR Ch. 2, Figure 3. General dimensions of the building are shown in 007N7334 “Power Block General Arrangement,” (Reference 9B-69).

9B.3.1.1 Structural Role and Safety Function

9B.3.1.1.1 Structural Role

The RWB houses and structurally supports the equipment associated with the Liquid Waste Management System (LWM), Solid Waste Management System (SWM), and OGS, which includes tanks, processing equipment, storage areas, and support facilities. Storage for dry active waste is also provided in the RWB. The Plant Chillers are located on the roof of the RWB. Work locations in the RWB are habitable and operable under all applicable plant states where operator actions are required.

9B.3.1.1.2 Safety Design Bases

Detailed SFRs for the RWB will be developed later when more detail is available for the Pre-Construction Safety Report.

The RWB is classified as Safety Class 3 (SC3) consistent with the highest-class component housed in the building.

The RWB is also classified as a BWRX-300 Seismic Category RW as indicated in PSR Ch. 3, Table 3-1, which meets the criterial for RW-IIa (High Hazard) in USNRC RG 1.143, “Design Guidance for Radioactive Waste Management Systems, Structures and Components Installed in Light-Water-Cooled Nuclear Power Plants,” (Reference 9B-70). Consequently, the Seismic Category RW RWB is designed to remain essentially elastic without any significant permanent deformation up to half of the DBE and satisfies RW-IIa design requirements in USNRC RG 1.143.

Due to its proximity to the RB, the RWB is evaluated for seismic and extreme wind interaction with the RB. The seismic interaction evaluation is based on the full DBE.

9B.3.1.2 Structural Description

The RWB is located next to the RB and between the TB and CB.

The RWB is a three-story structure consisting of a reinforced concrete exterior wall system with reinforced concrete slabs, beams, girders, and columns in the interior and a reinforced concrete deck on open web steel joists, supported by steel columns for the roof. It is anticipated that the structure is supported on a shallow reinforced concrete mat foundation however the foundation characteristics may vary depending on selected site conditions. The lateral force resisting system of the structure consists of concrete shear walls with concrete floors, and composite roof decking acting as diaphragms. The open top radwaste water storage tanks are of concrete construction with an alloy steel liner. Interior reinforced concrete walls are provided for radiological shielding of the LWM, the SWM and OGS.

9B.3.1.3 Structural Analysis and Design Basis

The RWB forms part of the combined model used for the SSI analyses of the integrated RB to account for the SSI and SSSI effects on the structure as described in PSR Ch. 3, Section 3.3.1.

The design of the RWB structure is performed following the linear elastic analysis methodology using a standalone FE model with a sufficient level of refinement to allow for the accurate estimation of design stresses and forces.

The RWB is designed in accordance with USNRC RG 1.143, which invokes ANSI/AISC N690 and ACI 349. Seismic analysis of the RWB is performed in accordance with ASCE/SEI 4 and ASCE/SEI 43.

The design of secondary structural elements (i.e., components and cladding, roof, other structures and building appurtenances) is site-specific and is per applicable local building code requirements.

9B.3.1.3.1 Applicable Codes, Standards and Specifications

The design, fabrication, construction, testing, and ISI of the RWB is performed in accordance with USNRC RG 1.143, which establishes the appropriate 10 CFR 50 requirements (including the provisions of GDCs 1, 2, and 4), following the regulatory guidance of NUREG-0800, SRP 3.8.4.

Analysis procedures for the RWB are in accordance with ASCE/SEI 4 and ASCE/SEI 43.

9B.3.1.3.2 Loads and Load Combinations

Loads and load combinations used in the design of the RWB are in accordance with 008N0279, “BWRX-300 Design Specification for Radwaste Building Structure,” (Reference 9B-71), with design earthquake and tornado loads applicable to RWB defined per the provisions of USNRC RG 1.143 and site-specific design basis hurricane wind load per ASCE/SEI 7. Environmental load combinations per 006N8282, “BWRX-300 Civil Structural Interaction Evaluation Criteria,” (Reference 9B-72) are considered for interaction evaluations of RWB with RB under DBE and design basis extreme wind conditions.

9B.3.1.3.3 Seismic Analysis Approach

A linear elastic analysis methodology, using a standalone FE model with a sufficient level of refinement to allow for the accurate estimation of design stresses and forces, is used for the seismic design and seismic interaction evaluation of the RWB. Structural damping values are per USNRC RG 1.61, “Damping Values for Seismic Design of Nuclear Power Plants,” (Reference 9B-73).

The ASCE/SEI 4 Two-Step analysis approach is used for the seismic qualification of the RWB structure where seismic demands in the form of RWB zero period accelerations or foundation acceleration response spectra are developed from the RB SSI analyses for responses of the RWB simplified dynamic model discussed in PSR Ch. 3, Section 3.3.1.3. A fixed base quasi-static analysis using the maximum nodal accelerations or response spectrum analysis using the input foundation response spectrum enveloped with the RWB design earthquake is then performed on the standalone refined FE model of the RWB. Resulting forces and stresses are used for the seismic design of the RWB. The demands for the RWB foundation design are taken directly from the RB SSSI analysis, with input corresponding to the RWB design earthquake.

The seismic demands for subsystem design and evaluation are developed based on In-Structure Response Spectra (ISRS), acceleration time histories and relative displacements calculated with the Response Level 1 (OBE) structural damping values. The use of models with higher (Response Level 2, DBE) damping values can be justified based on concrete cracking assumptions as discussed in PSR Ch. 3, Section 3.3.1.2.

9B.3.1.3.4 Building Interactions

The interaction evaluations of the RWB lateral force resisting system are performed for full DBE loads using the results of the seismic SSI analysis, and for extreme wind/tornado loads, whilst considering limited permanent deformations with expected minimal damage per ASCE/SEI 43Limit State LS-C. As mentioned in PSR Ch. 3, Section 3.3.1.3, a gap is maintained between the RB and the RWB to prevent any physical contact during a seismic or extreme wind event. Clear gaps also separate the RWB from the adjacent CB and TB.

9B.3.1.3.5 Concrete and Steel Design and Detailing

The RWB design follows the regulatory guidance of NUREG-0800, SRP 3.8.4 and USNRC RG 1.143. Design and detailing of the RWB structural steel is in accordance with ANSI/AISC N690 as supplemented by USNRC RG 1.243. Design and detailing of the RWB structural concrete is in accordance with ACI 349, as supplemented by USNRC RG 1.142, “Safety-Related Concrete Structures for Nuclear Power Plants (Other than Reactor Vessels and Containments),” (Reference 9B-74).

9B.3.1.3.6 Fire Protection

The RWB is designed to have a 3-hour fire resistance rating. Fire protection features for the RWB comply with requirements in RG 1.189. Details associated with fire protection system design for this structure are provided in PSR Ch. 9A, Section 9A.6.

9B.3.1.4 Materials

The materials used in the construction of the RWB are in accordance with ACI 349 for concrete structures and ANSI/AISC N690 for steel structures meeting USNRC RG 1.143 Table 1 guidance for Seismic Category RW SSCs.

Construction, fabrication, and installation of metallic liners, non-metallic liners, coatings, joint sealants, and waterstops for structures where leak tightness is critical (if any) are in line with the requirements of ASME BPVC Code Section III, Division 2 (Subarticle CC-2500 requirements are followed for liner materials).

9B.3.1.5 Interfaces with Other Equipment or Systems

The RWB houses process systems such as the LWM, SWM, a portion of the OGS including the charcoal adsorbers, with the exception of the condensate filters and demineralizers located in the TB, and other systems.

9B.3.1.6 System and Equipment Operation

Section is not applicable to the RWB.

9B.3.1.7 High level Construction Considerations

Construction requirements are defined on drawings and/or specifications. Construction planning and constructability reviews are performed concurrently with the structural design.

The RWB structure, being principally a reinforced concrete structure, is comprised of prefabricated and cast in-situ panels built using traditional building forming methods with self-consolidating concrete used to reduce cold joints. Other units within the RWB are installed as assemblies or modules wherever possible.

CDM 2015 Regulations will be applied as described in PSR Ch. 14.

9B.3.1.8 Instrumentation and Control

Section is not applicable to the RWB.

9B.3.1.9 Monitoring, Testing, Inspection and Maintenance

Through life monitoring, testing, inspection, and maintenance will be conducted in line with safety requirements, once defined. Refer to NEDC-34174P, "BWRX-300 UK GDA Ch. 11: Management of Radioactive Waste," (Reference 9B-75) for the testing and in-service requirements for process systems inside the RWB.

9B.3.1.10 Radiological Aspects

Design features for radiation protection in the RWB are discussed in PSR Ch. 12.

9B.3.1.11 Performance and Safety Evaluation

To meet its functional and performance requirements listed in Section 9B.3.1.1, the RWB is evaluated and designed for the design loads described in Section 9B.3.1.3.2 using the methods and design basis outlined in Section 9B.3.1.3.

For all design basis load cases, the RWB structure is designed to meet the acceptance criteria outlined in ANSI/AISC N690 as endorsed and modified by USNRC RG 1.243 and ACI 349 as endorsed and modified by USNRC RG 1.142, following the regulatory guidance of NUREG-0800, SRP 3.8.4.

DECs and beyond design basis requirements are not explicitly considered for the RWB, however the RWB will be inherently robust as it will conform to the requirements of ACI 349 and ANSI/AISC N690.

9B.3.2 Control Building

The CB houses the Main Control Room (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 Reactor Building.

The CB location is shown in PSR Ch. 2, Figure 3. General dimensions of the building are shown in 007N7334.

9B.3.2.1 Structural Role and Safety Function

9B.3.2.1.1 Structural Role

The CB houses and structurally supports the MCR, Distributed Control and Information System rooms, associated equipment, batteries, and Uninterruptible Power Supplies (UPSs), and the CB HVAC equipment. Work locations in the CB are habitable and operable under all applicable plant states where operator actions are required.

The qualified evacuation route from the MCR to the SCR (housed within the RB) protects personnel from the internal and external hazards generated from the Postulated Initiating Event (PIE) (that initiated evacuation of the MCR). As discussed in PSR Ch. 6 Section 6.6.1.1, an alternative evacuation route from the MCR to the SCR exists, for use in case of fire affecting the primary path. No additional hazard protection beyond the standards of the building(s) it passes through is provided for the alternative route.

9B.3.2.1.2 Safety Design Bases

Detailed SFRs for the CB will be developed later when more detail is available for the Pre-Construction Safety Report.

The CB is classified as Safety Class 2 (SC2) consistent with the highest-class component housed within the building and is a BWRX-300 Seismic Category 2 structure as indicated in PSR Ch. 3, Table 3-1. Seismic Category 2 applies to select Seismic Category Non-Seismic

(NS) civil structures that cannot be ruled out for interaction with BWRX-300 Seismic Category 1A/1B SSCs during DBE or extreme wind events (i.e., collapsing of NS structure causes the NS structure to strike a BWRX-300 Seismic Category 1A/1B SSC, or impair the integrity of the BWRX-300 Seismic Category 1A/1B SSC) due to proximity to BWRX-300 Seismic Category 1A structures.

Due to its proximity to the RB, the CB is evaluated for seismic and extreme wind interaction with the RB. The seismic interaction evaluation is based on the full DBE.

Prefabricated structures located to the south of the CB, are not Safety Classified and are Seismic Category NS structures.

To meet the Seismic Category 2 Interaction guidance of NUREG-0800,Section 3.7.2, “Seismic System Analysis,” (Reference 9B-76) Subsection II.8 and USNRC RG 1.29, “Seismic Design Classification for Nuclear Power Plants,” (Reference 9B-77), there is a requirement that under DBE or extreme wind events, no incapacitating injury of the MCR occupants shall occur, and safe egress for operators shall be ensured, requiring an assessment of the MCR and MCR to SCR primary evacuation route as described in 9B.3.2.3.1.

9B.3.2.2 Structural Description

The CB is located next to the RB, between the RWB and the Service Building.

The CB structure consists of a three-story building frame system with perimeter reinforced concrete shear wall, interior steel columns, beams/girders, and steel-concrete roof deck as a gravity load carrying system. It is anticipated that the CB is founded on a reinforced concrete mat foundation, however the foundation characteristics may vary depending on selected site conditions. The lateral force resisting system of the structure consists of reinforced concrete shear walls with concrete slabs acting as diaphragms.

The structural system of the MCR and the qualified evacuation route between the MCR and SCR may be integrated within part of the CB lateral force resisting system or may be an independent structural system within the CB envelope.

9B.3.2.3 Structural Analysis and Design Basis

It is considered that the CB analysis and design will be based on requirements in Eurocodes, and in particular Eurocode 2, “Design of concrete structures - General rules and rules for buildings, bridges and civil engineering structures,” (Reference 9B-78) for concrete parts and Eurocode 3, “Design of steel structures – Part 1-1: General rules and rules for buildings,” (Reference 9B-79) for steel parts. Consequence Class 3 should be considered for seismic design as per PD 6698, “Recommendations for the design of structures for earthquake resistance to BS EN 1998,” (Reference 9B-80) unless otherwise stated below, for seismic interaction analysis for example.

The CB forms part of the combined model used for the SSI analyses of the integrated RB to account for the SSI and SSSI effects on the structure as described in PSR Ch. 3, Section 3.3.1.

The design of the CB structure is performed following a linear elastic analysis methodology using a standalone FE model with a sufficient level of refinement to allow for the accurate estimation of design stresses and forces.

The design of secondary structural elements (i.e., components and cladding, roof, other structures and building appurtenances) is site-specific and shall be per applicable local building code requirements.

9B.3.2.3.1 Special Hardening Provision for Control Room

To ensure the safety of operators in the MCR, the MCR is surrounded by reinforced concrete shear walls. Per PSR Ch. 6, Section 6.6, operators are expected to remain in the MCR and safely operate the plant for most of the PIEs. For more details on the conditions that necessitate an evacuation of the MCR, and on travel routes provided to ensure safe egress of operators, refer to PSR Ch. 6, Section 6.6.

To ensure personnel safety, habitability and safe egress to SCR, and to meet plant level requirements and the regulatory guidance of NUREG-0800, SRP 3.7.2, Subsection II.8, USNRC RG 1.117, “Protection Against Extreme Wind Events and Missiles for Nuclear Power Plants,” (Reference 9B-81) and USNRC RG 1.29, the CB structure is designed not to collapse and result in incapacitating injury to the occupants of the MCR or prevent the egress of the MCR occupants to the SCR. To meet these requirements parts of the CB (at a minimum the MCR including the work control center and access control room, operations supervisor room, toilet room and MCR to SCR primary evacuation route) are:

- Hardened for protection from extreme wind (tornados and hurricanes) and perforation by extreme wind missiles. The extreme wind load is the load used for BWRX-300 Seismic Category 1A SSCs design and includes the full tornado (1E-7 per year return period) effect and hurricane missiles.
- Evaluated for full DBE with limited permanent deformations and with expected minimal damage in line with ASCE/SEI 43 Limit State LS-C.
- The MCR is also designed to meet the bullet resistant requirements of 10 CFR 73.55(e)(5). MCR doors are designed to remain functional and able to open during and after the design basis extreme environmental event.
- Unobstructed access of the egress route during and after the events which can initiate an evacuation of the MCR is also ensured by the design to secure the egress of MCR occupants to the SCR.

The areas/structures under this category are designed to remain functional before, during, and after the design extreme environmental events.

9B.3.2.3.2 Applicable Codes, Standards and Specifications

It is considered that reinforced concrete parts of the CB are designed in accordance with provisions of Eurocode 2.

It is considered that parts of the CB constructed with structural steel are designed in accordance with the provisions Eurocode 3.

The CB Standard Plant design developed in the U.S., using U.S. local codes and standards, is assessed using Eurocodes at site-specific stage in the UK.

ASCE/SEI 4 and ASCE/SEI 43 are used in the analysis of the lateral force resisting system to ensure that there are no issues with interaction between the CB and the RB.

9B.3.2.3.3 Loads and Load Combinations

Loads and load combinations used in the design of the CB are in accordance with the 008N0278, “BWRX-300 Design Specification for Control Building Structure,” (Reference 9B-82).

9B.3.2.3.4 Seismic Analysis Approach

The seismic design and seismic interaction evaluation of the CB structure are performed following the linear elastic analysis methodology using a standalone FE model with a sufficient level of refinement to allow for the accurate estimation of design stresses and forces.

PD 6698 describes requirements for determining earthquake loads for Consequence Class 3 structures.

For the hardened structural elements of the CB, see Section 9B.3.2.3.19B.3.2.3.1 for the seismic design basis.

9B.3.2.3.5 Building Interactions

The interaction evaluations of the CB lateral force resisting system are performed for full DBE loads using the results of the seismic SSI analysis, and for extreme wind/tornado loads, whilst considering limited permanent deformations with expected minimal damage per ASCE/SEI 43 Limit State LS-C. As mentioned PSR Ch. 3, Section 3.3.1.3, a gap is maintained between the RB and the CB to prevent any physical contact during a seismic or extreme wind event. Clear gaps also separate the CB from the adjacent RWB and Service Building.

9B.3.2.3.6 Concrete and Steel Design and Detailing

It is considered that design and detailing of the CB structural steel is in accordance Eurocode 2 (for structural concrete) and Eurocode 3 (for structural steel).

The CB Standard Plant design developed in the U.S. is assessed against Eurocodes at site-specific stage in the UK.

9B.3.2.3.7 Fire Protection

The CB is designed to have a 3-hour fire resistance rating. Fire protection features for the CB comply with requirements in RG 1.189. Details associated with fire protection system design for this structure are provided in PSR Ch. 9A, Section 9A.6.

9B.3.2.4 Materials

It is considered that concrete material for Seismic Category NS and Seismic Category 2 structures will conform to Eurocode 2.

It is considered that material properties of structural steel for other Seismic Category NS and Seismic Category 2 structures will conform to Eurocode 3.

9B.3.2.5 Interfaces with Other Equipment or Systems

The CB houses and structurally supports the MCR, distributed control and information system rooms, associated equipment, batteries and UPSs, and the CB HVAC equipment.

9B.3.2.6 System and Equipment Operation

Section is not applicable to the CB.

9B.3.2.7 High level Construction Considerations

Construction requirements are defined in drawings and/or specifications. Construction planning and constructability reviews are performed concurrently with the structural design.

Construction of the CB generally uses conventional building methods and does not involve significant structural modules. However, there are some structural elements within this building that are assumed as modules or assemblies. These include:

- Floor deck Assemblies for any elevated floor slabs present in the building
- Stair tower assemblies for all staircases present in the building

CDM 2015 Regulations will be applied as described in PSR Ch. 14

9B.3.2.8 Instrumentation and Control

Section is not applicable to the CB.

9B.3.2.9 Monitoring, Testing, Inspection and Maintenance

Through life monitoring, testing, inspection, and maintenance will be conducted in line with safety requirements, once defined.

9B.3.2.10 Radiological Aspects

Design features for radiation protection in the CB are discussed in PSR Ch. 12. Design features for habitability of the MCR are described in PSR Ch. 6, Section 6.6.

9B.3.2.11 Performance and Safety Evaluation

To meet its functional and performance requirements listed in Section 9B.3.2.1, the CB is evaluated and designed for the design loads described in Section 9B.3.2.3.3 using the methods and design basis outlined in Section 9B.3.2.3.

The design of the CB Seismic Category 2 SSCs shall be in accordance with the requirements of the PD 6698 for Consequence Class 3 structures, the associated Eurocode design requirements and 008N0278. This also applies to the Seismic Category NS detached prefabricated/modularized equipment structures associated with the CB.

Design of the Standard Plant Lateral Force Resisting System of the CB is based on Standard Plant loads as described in 008N0278, if not governed by site-specific BWRX-300 Seismic Category 1A demands (as per the special hardening requirements). As site-specific values become available, the design of the Lateral Force Resisting System is re-evaluated to confirm that the design is conservative. For the Seismic Category 2 Lateral Force Resisting System, to account for the inelastic response, the DBE demands obtained from the results of linear elastic seismic response analysis are reduced, based on the structural system, using Limit State LS-C inelastic energy absorption factors provided in Table 5-1 of ASCE/SEI 43. The reduced DBE demands are combined with non-seismic demands, as shown in Table 7-1 of 008N0278 to evaluate the structural integrity of the Lateral Force Resisting System of the CB per governing nuclear design codes ACI 349 and ANSI/AISC N690.

DECs and beyond design basis robustness requirements are not explicitly considered for the CB.

9B.3.3 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 demineralizers, bridge crane and other systems.

The TB location is shown in PSR Ch. 2, Figure 3. General dimensions of the building are shown in 007N7334.

9B.3.3.1 Structural Role and Safety Function

9B.3.3.1.1 Structural Role

The role of the turbine building is to house and protect the turbine, generator, condenser, some of the off-gas system and other safety systems related to radioactive steam and water.

9B.3.3.1.2 Safety Design Bases

Detailed SFRs for the TB will be developed later when more detail is available for the Pre-Construction Safety Report.

The TB is classified as SC2 consistent with the highest-class component housed within the building and is a BWRX-300 Seismic Category 2 structure similar to the CB and as indicated in PSR Ch. 3, Table 3-1.

Due to its proximity to the RB, the TB is evaluated for seismic interaction and extreme wind interaction with the RB. The seismic interaction evaluation is based on the full DBE.

9B.3.3.2 Structural Description

The TB is located next to the RB and RWB.

The TB structure is divided into three separate structural systems:

- The TB shell structure, which consists of a steel frame system with steel columns, beams/girders, roof bar joists, and floor/roof decks as gravity load carrying systems. The lateral force resisting system consists of braced frames and floor/roof decks as diaphragms.
- The TB Shield Wall Area consists of a perimeter reinforced concrete shear wall. The shear wall provides both radiation shielding and gravity/lateral structural capacity for the Shield Wall Area. The Shield Wall Area consists of a platform frame system around the turbine cycle equipment and a western portion of reinforced concrete. The platform frame system consists of steel columns, beams/girders, and floor decks as a gravity load bearing system around the turbine generator pedestal. The turbine cycle equipment area platform system ties into the reinforced concrete portion on the west side of the Shear Wall Area. This reinforced concrete portion consists of concrete floors and walls. The lateral force resisting system consists of reinforced concrete shear walls and floor levels as diaphragms.
- The reinforced concrete pedestal supporting the turbine, generator, and exciter within the TB shell structure and the shielding walls area. This turbine generator foundation is structurally isolated from both the shielding walls and the TB shell structure for vibration control. However, its dominant dynamic characteristics are included in the SSSI model.

It is anticipated that the TB is founded on a reinforced concrete mat foundation, however the foundation characteristics may vary depending on selected site conditions.

9B.3.3.3 Structural Analysis and Design Basis

It is considered that TB analysis and design will be based on requirements in Eurocodes, and in particular Eurocode 2 for concrete parts and Eurocode 3 for steel parts. Consequence Class 3 should be considered for seismic design as per PD 6698 unless otherwise stated below, for seismic interaction analysis for example.

The TB forms part of the combined model used for the SSI analyses of the integrated RB to account for the SSI and SSSI effects on the structure as described in PSR Ch. 3, Section 3.3.1.

The design of the TB structure is performed following a linear elastic analysis methodology using a standalone FE model with a sufficient level of refinement to allow for the accurate estimation of design stresses and forces.

Design of secondary structural elements (i.e., components and cladding, roof, other structures and building appurtenances) is site-specific and shall be per local building code requirements.

The turbine generator foundation is sized and designed to additionally meet the applicable static and dynamic performance requirements by the turbine generator manufacturer, and to have adequate structural integrity and stability to withstand applicable loads and associated load combinations imposed during the service life of the plant.

9B.3.3.3.1 Applicable Codes, Standards and Specifications

It is considered that reinforced concrete parts of the TB are designed in accordance with provisions of Eurocode 2.

It is considered that parts of the TB constructed with structural steel are designed in accordance with the provisions Eurocode 3.

The TB Standard Plant design developed in the U.S., using U.S. local codes and standards, is assessed using Eurocodes at site-specific stage in the UK.

ASCE/SEI 4 and ASCE/SEI 43 are used in the analysis of the lateral force resisting system to ensure that there are no issues with interaction between the TB and the RB.

9B.3.3.3.2 Loads and Load Combinations

Loads and load combinations used in the design of the TB are in accordance with the 008N0277, “BWRX-300 Design Specification for Turbine Building Structure,” (Reference 9B-83).

9B.3.3.3.3 Seismic Analysis Approach

The seismic design and seismic interaction evaluation of the TB structure are performed following the linear elastic analysis methodology using a standalone FE model with a sufficient level of refinement to allow for the accurate estimation of design stresses and forces.

PD 6698 describes requirements for determining earthquake loads for Consequence Class 3 structures.

9B.3.3.3.4 Building Interactions

The interaction evaluations of the TB lateral force resisting system are performed for full DBE loads using the results of the seismic SSI analysis, and for extreme wind/tornado loads, whilst considering limited permanent deformations with expected minimal damage per ASCE/SEI 43 Limit State LS-C. As mentioned PSR Ch. 3, Section 3.3.1.3, a gap is maintained between the RB and the TB to prevent any physical contact during a seismic or extreme wind event. A clear gap also separates the TB from the adjacent RWB.

9B.3.3.3.5 Concrete and Steel Design and Detailing

It is considered that design and detailing of the TB structural steel is in accordance with Eurocode 2 (for structural concrete) and Eurocode 3 (for structural steel).

The TB Standard Plant design developed in the U.S. is assessed against Eurocodes at site- specific stage in the UK.

9B.3.3.3.6 Fire Protection

The TB is designed to have a 3-hour fire resistance rating. Fire protection features for the TB comply with requirements in RG 1.189. Details associated with fire protection system design for this structure are provided in PSR Ch. 9A, Section 9A.6.

9B.3.3.4 Materials

It is considered that concrete material for Seismic Category NS and Seismic Category 2 structures will conform to Eurocode 2.

It is considered that material properties of structural steel for other Seismic Category NS and Seismic Category 2 structures will conform to Eurocode 3.

9B.3.3.5 Interfaces with Other Equipment or Systems

The TB encloses the turbine generator, main condenser, condensate and feedwater systems, condensate purification system, off-gas catalytic recombiner, cooling condenser, refrigerant dryer, turbine generator support system and bridge crane.

9B.3.3.6 System and Equipment Operation

Section is not applicable to the TB.

9B.3.3.7 High level Construction Considerations

Construction requirements are defined on drawings and/or specifications. Construction planning and constructability reviews are performed concurrently with the structural design.

Modularization is considered for many elements of the TB including:

- The Turbine Pedestal components
- Turbine Building Radiation Shielding Wall
- Floor Deck Assemblies used on various parts of the TB
- Access Platform Assemblies
- Stair towers

CDM 2015 Regulations will be applied as described in PSR Ch. 14.

9B.3.3.8 Instrumentation and Control

Section is not applicable to the TB.

9B.3.3.9 Monitoring, Testing, Inspection and Maintenance

Through life monitoring, testing, inspection, and maintenance will be conducted in line with safety requirements, once defined.

9B.3.3.10 Radiological Aspects

Radiological aspects of the TB are discussed in PSR Ch. 12.

9B.3.3.11 Performance and Safety Evaluation

To meet its functional and performance requirements listed in Section 9B.3.3.1, the TB is evaluated and designed for the design loads described in Section 9B.3.3.3.2 using the methods and design basis outlined in Section 9B.3.3.3.

The design of the TB Seismic Category 2 SSCs shall be in accordance with the requirements of the PD 6698 for Consequence Class 3 structures, the associated Eurocode design requirements, and 008N0277.

Design of the Standard Plant Lateral Force Resisting System of the TB is based on Standard Plant loads as described in 008N0277. As site-specific values become available, the design of the Lateral Force Resisting System is re-evaluated to confirm that the design is conservative. For the Seismic Category 2 Lateral Force Resisting System, to account for the inelastic response, the DBE demands obtained from the results of linear elastic seismic response analysis are reduced, based on the structural system, using Limit State LS-C inelastic energy absorption factors provided in Table 5-1 of ASCE/SEI 43. The reduced DBE demands are combined with non-seismic demands, as shown in Table 7-1 of 008N0277 to evaluate the structural integrity of the Lateral Force Resisting System of the TB per governing nuclear design codes ACI 349 and ANSI/AISC N690.

DECs and beyond design basis requirements are not explicitly considered for the TB.

9B.3.4 Service Building, Reactor Auxiliary Structures and Fire Water storage Tank and Pump Enclosure

The Service Building, Reactor Auxiliary Structures and Fire Water Storage Tank and Pump Enclosure are of lower maturity currently; therefore, the below information is generic and will be refined for each individual structure as the design matures.

For the locations of these structures, refer to PSR Ch. 2, Figure 3. General dimensions of the buildings are shown in 007N7334.

9B.3.4.1 Structural Role and Safety Function

9B.3.4.1.1 Structural Role

The structural role of each building is described in each building's Design Specification.

9B.3.4.1.2 Safety Design Bases

Detailed SFRs for the structures will be developed later when more detail is available for the Pre-Construction Safety Report.

The Service Building and Reactor Auxiliary Structures are classified as SC2 consistent with the highest-class component housed within the building and are BWRX-300 Seismic Category 2 structures, similar to the CB and TB and as indicated in PSR Ch. 3, Table 3-1.

Due to their proximity to the RB, the Service Building and Reactor Auxiliary Structures are evaluated for seismic and extreme wind interaction with the RB. The seismic interaction evaluation is based on the full DBE.

The Fire Water Storage Tank and Pump Enclosure is classified SC3 consistent with the highest-class component housed within the building and is a BWRX-300 Seismic Category NS structure. As such the building is designed per local building codes.

9B.3.4.2 Structural Description

The Service Building is a rectangular three-story structure located next to the RB and CB and is separated from the structures by a seismic gap. The Service Building is not part of the Radiologically Controlled Area.

The Reactor Auxiliary Structures are comprised of the four arc sectors around the RB. Their design will be reviewed further as the BWRX-300 design progresses. These structures are associated with the Service Building.

No information is currently available on the structural description of the Fire Water Storage Tank and Pump Enclosure.

9B.3.4.3 Structural Analysis and Design Basis

It is considered that the analysis and design of the Service Building, Reactor Auxiliary Structures and Fire Water Storage Tank and Pump Enclosure will be based on requirements in Eurocodes, and in particular Eurocode 2 for concrete parts and Eurocode 3 for steel parts. Consequence Class 3 should be considered for seismic design as per PD 6698 unless otherwise stated below, for seismic interaction analysis for example.

The design of these structures is performed following a linear elastic analysis methodology using a standalone FE model with a sufficient level of refinement to allow for the accurate estimation of design stresses and forces.

Design of secondary structural elements (i.e., components and cladding, roof, other structures and building appurtenances) is site-specific and shall be per local building code requirements.

9B.3.4.3.1 Applicable Codes, Standards and Specifications

It is considered that reinforced concrete parts of the structures are designed in accordance with provisions of Eurocode 2.

It is considered that parts of the structures constructed with structural steel are designed in accordance with the provisions Eurocode 3.

The Standard Plant design of the structures developed in the U.S., using U.S. local codes and standards, is assessed using Eurocodes at site-specific stage in the UK.

ASCE/SEI 4 and ASCE/SEI 43 are used in the analysis of the lateral force resisting system of the Service Building and Reactor Auxiliary Structures to ensure that there are no issues with interaction between the structures and the RB.

9B.3.4.3.2 Loads and Load Combinations

Loads and load combinations used in the design of the structures will be in accordance with the building specifications for each structure, once available.

9B.3.4.3.3 Seismic Analysis Approach

The Fire Water Storage Tank and Pump Enclosure is not seismically designed.

The seismic design and seismic interaction evaluation of the Service Building and Reactor Auxiliary Structures are performed following the linear elastic analysis methodology using a standalone FE model with a sufficient level of refinement to allow for the accurate estimation of design stresses and forces.

PD 6698 describes requirements for determining earthquake loads for Consequence Class 3 structures.

9B.3.4.3.4 Building Interactions

The interaction evaluations of the Service Building and Reactor Auxiliary Structures are performed for full DBE loads using the results of the seismic SSI analysis, and for extreme wind/tornado loads, whilst considering limited permanent deformations with expected minimal damage per ASCE/SEI 43 Limit State LS-C. As mentioned PSR Ch. 3, Section 3.3.1.3, a gap is maintained between the RB and the structures to prevent any physical contact during a seismic or extreme wind event. Clear gaps also separate the structures from the adjacent RWB, CB and TB.

9B.3.4.3.5 Concrete and Steel Design and Detailing

9B.3.4.3.6 Fire Protection

It is considered that design and detailing of the structural steel is in accordance Eurocode 2 (for structural concrete) and Eurocode 3 (for structural steel).

The Service Building and Reactor Auxiliary Structures are designed to have a 3-hour fire resistance rating. Fire protection features comply with requirements in RG 1.189. Details associated with fire protection system design for the structures are provided in PSR Ch. 9A, Section 9A.6.

The fire rating for the Fire Water Storage Tank and Pump Enclosure will be provided later when more detail is available for the Pre-Construction Safety Report.

9B.3.4.4 Materials

It is considered that concrete material for Seismic Category NS and Seismic Category 2 structures will conform to Eurocode 2.

It is considered that material properties of structural steel for other Seismic Category NS and Seismic Category 2 structures will conform to Eurocode 3.

9B.3.4.5 Interfaces with Other Equipment or Systems

Section is to be completed later when more detail is available for the Pre-Construction Safety Report.

9B.3.4.6 System and Equipment Operation

Section is not applicable to the Service Building, Reactor Auxiliary Structures and Fire Water Storage Tank and Pump Enclosure.

9B.3.4.7 High Level Construction Considerations

Construction requirements are defined on drawings and/or specifications. Construction planning and constructability reviews are performed concurrently with the structural design.

Modularization and prefabrication are deployed wherever possible to expedite construction durations and improve overall quality.

CDM 2015 Regulations will be applied as described in PSR Ch. 14.

9B.3.4.8 Instrumentation and Control

Section is not applicable to the Service Building, Reactor Auxiliary Structures and Fire Water Storage Tank and Pump Enclosure.

9B.3.4.9 Monitoring, Testing, Inspection and Maintenance

Through life monitoring, testing, inspection, and maintenance will be conducted in line with safety requirements, once defined.

9B.3.4.10 Radiological Aspects

Radiological aspects of the Service Building are discussed in PSR Ch. 12, There are no radiological sources in the Reactor Auxiliary Structures and Fire Water Storage Tank and Pump Enclosure.

9B.3.4.11 Performance and Safety Evaluation

To meet their functional and performance requirements listed in Section 9B.3.4.1, the Service Building, Reactor Auxiliary Structures and Fire Water Storage Tank and Pump Enclosure are evaluated and designed for the design loads described in Section 9B.3.4.3.2 using the methods and design basis outlined in Section 9B.3.4.3.

It is considered that design of the Service Building, Reactor Auxiliary Structures and Fire Water Storage Tank and Pump Enclosure shall be in accordance with the requirements of PD 6698 for Consequence Class 3 structures and 005N9341, “BWRX-300 General Civil Structural Design Criteria,” (Reference 9B-84). Design of the Standard Plant lateral force resisting system of the Service Building and Reactor Auxiliary Structures is based on Standard Plant loads which will be described in each building’s specification. As site-specific values become available, the design of the lateral force resisting system is re-evaluated to confirm that the design is conservative. For the Seismic Category 2 lateral force resisting system, to account for the inelastic response, the DBE demands obtained from the results of linear elastic seismic response analysis are reduced, based on the structural system, using LS-C inelastic energy absorption factors provided in Table 5-1 of ASCE/SEI 43. The reduced DBE demands are combined with non-seismic demands to evaluate the structural integrity of the Service Building and Reactor Auxiliary Structures lateral force resisting system per governing nuclear design codes ACI 349 and ANSI/AISC N690.

DECs and beyond design basis requirements are not explicitly considered for the Service Building, Reactor Auxiliary Structures and Fire Water Storage Tank and Pump Enclosure.

Table 9B-1: Stability Requirements for Power Block Foundations

Load Combination	Overturning	Sliding	Flotation
$D + H + W$	1.5	1.5	
$D + H + E'$	1.1	1.1	
$D + H + W_t$	1.1	1.1	
$D + F'$			1.1

where
D = Dead Load, W = Wind
H = Lateral soil pressure, E' = Design Basis Earthquake
W_t = Tornado or Hurricane Wind
F' = Buoyant forces of design basis flood

Table 9B-2: Load Combinations and Load Factors for Steel-Plate Composite Containment Vessel

Load Category	Combination #	Loads																		Load Condition
		D	F	L	H	P_t	T_t	P_v/ P_o	R_o	T_o	W	E_s	W_t	P_a	R_a	T_a	Y_j/R_{rj}	Y_m/ R_{rm}	Y_r/R_{rr}	
Test	SCCV-1	1.0	1.0	1.0	1.0	1.0	1.0													Service
Construction	SCCV-2	1.0	1.0	1.0	1.0					1.0	1.0									
Normal	SCCV-3	1.0	1.0	1.0	1.0			1.0	1.0	1.0										
Severe Environmental	SCCV-4	1.0	1.0	1.30	1.30			1.0	1.0	1.0	1.5									Factored
Extreme Environmental	SCCV-5	1.0	1.0	1.0	1.0			1.0	1.0	1.0		1.0								
	SCCV-6	1.0	1.0	1.0	1.0			1.0	1.0	1.0			1.0							
Abnormal	SCCV-7	1.0	1.0	1.0	1.0									1.5	1.0	1.0				
	SCCV-8	1.0	1.0	1.0	1.0									1.0	1.25	1.0				
	SCCV-9	1.0	1.0	1.0	1.0									1.25	1.0	1.0				
Abnormal / Extreme Environmental	SCCV-10	1.0	1.0	1.0	1.0						1.25			1.25	1.0	1.0				
	SCCV-11	1.0	1.0	1.0	1.0						-	1.0		1.0	1.0	1.0	1.0	1.0	1.0	

Notes:

1. In the Standard design, the load combination involving post-LOCA flooding condition with OBE set at 1/3 of the OBE is eliminated in accordance with NUREG-0800 SRP 3.8.1. This load combination is considered bounded by SCCV-11 because the maximum hydrostatic pressure produced by the post-LOCA flooding is less significant than the accident LOCA pressure.

D = Dead Loads

F = Hydrostatic Pressure Loads

L = Live Loads

H = Soil and Underground Water Pressure Loads

P_t = Test Pressure Loads

T_t = Test Thermal Loads

P_v/P_o = Pressure Variant Load

R_o = Normal Operating Pipe Reaction Loads

T_o = Normal Operating Thermal Loads

W = Wind Loads

E_s = Seismic Loads (generated by DBE)

W_t = Tornado or Hurricane Loads

P_a = Accident Pressure Loads

R_a = Accident Pipe Reaction Loads

T_a = Accident Thermal Loads

Y_j/R_{rj} = Jet Impingement Loads

Y_m / R_{rm} = Missile Impact Loads

Y_r / R_{rr} = High Energy Pipe Rupture

Table 9B-3: Load Combinations, Load Factors and Acceptance Criteria for Non-Containment Seismic Category 1A Steel-Plate Composite Modules and Steel Structures

(Adopted from ANSI/AISC N690-18 and USNRC RG 1.243)

Load Category	Combination #	Loads(2)																				
		D	F	L	H	L_r	R	S	C	R_o	P_o	T_o	W	E_s	W_t(7)(8)(10)	S_x	P_a(3)	R_a	T_a(3)	Y_j(4)	Y_m(4)	Y_r(4)
Normal	RB-1	1.4	1.4						1.0	1.4	1.0	1.0										
	RB-2.1	1.2	1.2	1.6	1.6	0.5			1.4	1.2	1.2	1.2										
	RB-2.2	1.2	1.2	1.6	1.6		0.5		1.4	1.2	1.2	1.2										
	RB-2.3	1.2	1.2	1.6	1.6			0.5	1.4	1.2	1.2	1.2										
	RB-3.1	1.2	1.2	0.8	0.8	1.6			1.4	1.2	1.2	1.2										
	RB-3.2	1.2	1.2	0.8	0.8		1.6		1.4	1.2	1.2	1.2										
	RB-3.3	1.2	1.2	0.8	0.8			1.6	1.4	1.2	1.2	1.2										
	RB-2/3	1.2	1.2	1.6	1.6			1.6	1.4	1.2	1.2	1.2										
Severe Environmental	RB-4.1	1.2	1.2	1.6	1.6	0.5			1.0	1.2	1.0	1.0	1.0									
	RB-4.2	1.2	1.2	1.6	1.6		0.5		1.0	1.2	1.0	1.0	1.0									
	RB-4.3	1.2	1.2	1.6	1.6			0.5	1.0	1.2	1.0	1.0	1.0									
Extreme Environmental	RB-5	1.0	1.0	0.8	1.0				1.0	1.0	1.0	1.0		1.0								
	RB-6	1.0	1.0	0.8	1.0					1.0	1.0	1.0			1.0							
	RB-7	1.0	1.0	0.8	1.0					1.0	1.0	1.0				1.0						
Abnormal	RB-8	1.0	1.0	0.8	1.0				1.0								1.4	1.0	1.0			
	RB-9	1.0	1.0	0.8	1.0									0.7			1.0	1.0	1.0	1.0	1.0	1.0

D = Dead Loads, including settlements, F = Hydrostatic Pressure Loads, L = Live Loads, H = Soil and Underground Water Pressure Loads, L_r = Roof Live Loads, R = Rain Loads, S = Snow Load, C = Crane Loads, R_o = Normal Operating Pipe Reactions, P_o = Normal Operating Pressure Loads, T_o = Normal Operating Thermal Loads, W = Wind Loads, E_s = Seismic Loads (Generated by DBE), W_t = Tornado or Hurricane Wind Loads, S_x = Extreme Snow Loads, P_a = Accidental Pressure Loads, R_a = Accidental Piping Reactions, T_a = Accidental Thermal Loads, Y_j = Jet Impingement Loads from a ruptured high-energy pipe, Y_m = Missile Impact Loads from a ruptured high-energy pipe, Y_r = High Energy Pipe Rupture

Notes:

1. As per NUREG-800, SRP 3.8.1, "Where post-Loss-of-Coolant Accident (LOCA) flooding is a design consideration for the plant, the load combination in the ASME Code containing LOCA flooding along with OBE should be considered. Where post-LOCA flooding is combined with the OBE set at one-third or less of the DBE for the plant, this load combination may be eliminated provided the load combination is shown to be less severe than one of the other load combinations." As per 005N9341, the OBE is set to less than or equal 1/3 the DBE in the Standard Plant design. Also, the maximum hydrostatic pressure produced by internal flood Load is negligible compared to the accident LOCA pressure. Therefore, post-LOCA flooding does not need to be considered in the structural evaluation of the SCCV.
2. For any load combination, if the dead load acts to stabilize the structure against the destabilizing effects of lateral force or uplift, the load factor on dead load shall be 0.90. For other loads acting to stabilize the structure against the destabilizing effects, if it can be demonstrated that the load is always present or occurs simultaneously with the other loads, the corresponding coefficient of that load shall be taken as 0.9, otherwise, the coefficient for that load is taken as zero. F shall be treated in the same manner as D, and H shall be treated in the same manner as L when stability evaluations are performed.
3. If P_a and T_a are time-dependent loads, their effects are superimposed accordingly.

4. Maximum value includes an appropriate dynamic load factor will be used unless an appropriate time-history analysis is performed to justify otherwise.
5. Dead Load includes settlements.
6. The effects of LOCA dynamic loads that originate inside the containment will be considered as applicable.
7. The normal roof snow load is considered in the normal load combinations. For extreme snow load, use load combination RB-6, replacing tornado load W_t with extreme snow load S_x.
8. The normal rain load is considered in the normal load combinations. For extreme rain load (under PMP event), use load combination RB-6, replacing tornado load W_t, with extreme rain load R_x.
9. The 0.7 factor for E_s will be combined absolutely with the accident loads. In lieu of this requirement, a factor of 1.0 for E_s may be combined with the accident loads by square root of the sum of the squares, if all provisions of ANSI/AISC N690, Commentary Section NB2.5, Paragraph 4, are satisfied.
10. The design for loads due to accidental explosions, accidental vehicle impacts, extreme snow load, and small aircraft impacts use load combination RB-6 with those loads in lieu of W_t, as well as further guidance provided in Appendix N9 to ANSI/AISC N690.

Table 9B-4: Load Combinations and Load Factors for Containment Closure Head

Service Level	Load Combination									
	D	**L**	**E_s**	**T_o**	**T_a**	**T_t**	**P_o**	**P_a**	**P_t**	**R_r**
Test Condition	1	1				1			1	
Design Condition	1	1			1			1		
Level A	1	1		1			1			
	1	1			1			1		
Level C	1	1	1		1			1		
	1	1	1	1			1			
Level D	1	1	1		1			1		1

Notes:

- Based on NUREG-0800, SRP 3.8.2 and USNRC RG 1.57
- D = Dead Loads

  L = Live Loads

  E_s = Seismic Loads (generated by DBE)

  T_o = Normal Operating Thermal Loads

  T_a = Accident Thermal Loads

  T_t = Test Thermal Loads

  P_o = Normal Operating Pressure Loads including Pressure Variant Loads (P_v)

  P_a = Accident Pressure Loads

  P_t = Test Pressure Loads

  R_r = Local Effects on Containment due to LOCA (R_{rj} + R_{rm})

  P_{g1} = Maximum Pressure due to 100% Cladding-Coolant Reaction

  P_{g2} = Maximum Pressure due to Hydrogen Burning

  P_{g3} = Maximum Pressure due to Post-Accident Inerting
- As per USNRC RG 1.216, combustible gas load cases (P_{g1}, P_{g2}, and P_{g3}) and associated load combinations are associated with DECs and are currently not presented in this table

Table 9B-5: Load Combinations and Load Factors for Other Class MC Components

Service Level	Load Combination												
	D	**L**	**E_s**	**T_o**	**T_a**	**T_t**	**P_o**	**P_a**	**P_t**	**R_r**	**R_o**	**R_a**	H_a
Test Condition	1	1				1			1				
Design Condition	1	1			1			1				1	
Level A	1	1		1			1				1		
	1	1			1			1				1	
Level C	1	1	1		1			1				1	
	1	1	1	1			1				1		
Level D	1	1	1		1			1		1		1	
Post-flooding Condition	1	1	1[2]										1

Notes:

(1) Based on NUREG-0800, SRP 3.8.2 and USNRC RG 1.57

(2) OBE consideration only for post-flooding condition and cyclic loading considerations, in accordance with NUREG-0800, SRP 3.8.2

(3) D = Dead Loads
L = Live Loads
E_s = Seismic Loads (generated by DBE)
T_o = Normal Operating Thermal Loads
T_a = Accident Thermal Loads
T_t = Test Thermal Loads
P_o = Normal Operating Pressure Loads including Pressure Variant Loads (Pv)
P_a = Accident Pressure Loads
P_t = Test Pressure Loads
R_r = Local Effects on Containment due to LOCA (Rrr + Rrj + Rrm)
R_o = Normal Operating Pipe Reaction Loads
R_a = Accident Pipe Reaction Loads
H_a = Internal Flooding Loads
P_{g1} = Maximum Pressure due to 100% Cladding-Coolant Reaction
P_{g2} = Maximum Pressure due to Hydrogen Burning
P_{g3} = Maximum Pressure due to Post-Accident Inerting

(4) As per USNRC RG 1.216, combustible gas load cases (P_{g1}, P_{g2}, and P_{g3}) and associated load combinations are associated with DECs and are currently not presented in this table

Table 9B-6: Acceptance Criteria for SCCV

(a) Allowable Stress/Strain Limits for Factored Loads

Material	Force Classification	Type of Force Action	Criteria for Factored Loads	
			Stress Limit	Strain Limit if any
Concrete	Primary	Membrane	$0.60f_c'$	-
		Membrane + Bending (2)	$0.75f_c'$	-
	Primary + Secondary (1)	Membrane	$0.75f_c'$	-
		Membrane + Bending (2)	$0.85f_c'$ (3)	-
Steel Plates	Primary	Membrane or Membrane + Bending (2)	$0.90F_y$	-
	Primary + Secondary (1)	Membrane or Membrane + Bending (2)	-	$2\varepsilon_y$ (4)

Notes:

(1) The primary portion of this calculated stress shall not exceed the allowable stress applicable when primary stress acts alone.

(2) The membrane portion of this calculated stress shall not exceed the allowable stress applicable when membrane stress acts alone.

(3) The maximum allowable primary-plus-secondary membrane and bending compressive stress of $0.85f'c$ corresponds to a limiting strain of 0.002 in./in (0.002 mm/mm). The concrete allowable stresses for factored compression loads shall be reduced, if necessary, to maintain structural stability per Subparagraph CC-3421.1 of ASME BPVC, Section III, Division 2.

(4) Limit for mechanical (net) strain due to primary forces, which is calculated by subtracting strain induced by secondary force (e.g., thermal strain) from the total strain as per Sub subparagraph CC-3422.1(e) of ASME BPVC, Section III, Division 2. For analysis purposes, ϵ_y is defined as the yield stress divided by Young's modulus.

(b) Allowable Stresses for Service Loads

Material	Force Classification	Type of Force Action	Criteria for Service Loads
			Stress Limit
Concrete	Primary	Membrane	$0.30f_c'$
		Membrane + Bending (2)	$0.45f_c'$
	Primary + Secondary (1)	Membrane	$0.45f_c'$
		Membrane + Bending (2)	$0.60f_c'$
Steel Plates	Primary	Membrane or Membrane + Bending (2)	$0.50F_y$ (3) (4)
	Primary + Secondary (1)	Membrane or Membrane + Bending (2)	$0.67F_y$ (3) (4

Notes:

(1) The primary portion of this calculated stress shall not exceed the allowable stress applicable when primary stress acts alone.

(2) The membrane portion of this calculated stress shall not exceed the allowable stress applicable when membrane stress acts alone.

(3) Where the steel plates are under tension stress, the stress limit may be increased by 50%, with an upper limit of 0.9Fy when the temporary pressure loads during the test condition are combined with other loads in the load combination.

(4) Where the steel plates are under compression stress, the stress limit may be increased by 33%, with an upper limit of 0.9Fy when the temporary pressure loads during the test condition are combined with other loads in the load combination.

Table 9B-7: Acceptance Criteria for Containment Closure Head

Service Level	Acceptance Criteria*1			
	P_m	P_L	P_L+P_b *2	P_L+P_b+Q
Test Condition	0.8 S_y	1.15 S_y	1.15 S_y	N/A*3
Design Condition	1.0 S_{mc}	1.5 S_{mc}	1.5 S_{mc}	N/A*3
Level A	1.0 S_{mc}	1.5 S_{mc}	1.5 S_{mc}	3.0 S_m
Level C	1.2 S_{mc} or*4 1.0 S_y	1.8 S_{mc} or*4 1.5 S_y	1.8 S_{mc} or*4 1.5 S_y	N/A*3
Level D	S_f	1.5 S_f	1.5 S_f	N/A*3

P_m = primary stress: general membrane.

P_L = primary stress: local membrane.

P_b = primary stress: bending.

Q = secondary stress: membrane plus bending.

S_y = material's yield strength at temperature as in ASME BPVC Section II, Part D, Table Y-1.

S_m = allowable stress intensity Sm is the value given in ASME BPVC Section II Part D, Subpart 1, Tables 2A and 2B.

S_{mc} = allowable stress intensity S_{mc} is 1.1 times the S listed in ASME BPVC Section II Part D, Subpart 1, Tables 1A and 1B, except S_{mc} is not to exceed 90% of the material's yield strength at temperature shown in ASME BPVC Section II, Part D, Subpart 1, Tables Y-1.

S_f = 85% of the general primary membrane allowable permitted in Mandatory Appendix XXVII, ASME BPVC Code Section III. In the application of Appendix XXVII, S_m, if applicable, is as specified in NE-3112.4(a)(1).

*1: Acceptance Criteria is defined by ASME BPVC, Subsection NE, Subparagraph NE-3221.1 through 3221.4.

*2: Values shown are for a rectangular section. See ASME BPVC, Subsection NE, Subsubparagraph NE-3221.3(d) for other than a solid rectangular section.

*3: N/A = Not applicable. No evaluation required.

*4: The larger of the two values listed is chosen as a limit load.

Table 9B-8: Acceptance Criteria for Other MC Components

Service Level	Acceptance Criteria*1			
	P_m	P_L	P_L+P_b *2	P_L+P_b+Q
Test Condition	0.8 S_y	1.15 S_y	1.15 S_y	N/A*3
Design Condition	1.0 S_{mc}	1.5 S_{mc}	1.5 S_{mc}	N/A*3
Level A	1.0 S_{mc}	1.5 S_{mc}	1.5 S_{mc}	3.0 S_m
Level C	1.2 S_{mc} or*4 1.0 S_y	1.8 S_{mc} or*4 1.5S_y	1.8 S_{mc} or*4 1.5 S_y	N/A*3
Level D	S_f	1.5 S_f	1.5 S_f	N/A*3
Post-flooding Condition	1.2 S_{mc} or*4 1.0 S_y	1.8 S_{mc} or*4 1.5 S_y	1.8 S_{mc} or*4 1.5 S_y	3.0 S_m

P_m = primary stress: general membrane.

P_L = primary stress: local membrane.

P_b = primary stress: bending.

Q = secondary stress: membrane plus bending.

S_y = material's yield strength at temperature as in ASME BPVC Section II, Part D, Table Y-1.

S_m = allowable stress intensity S_m is the value given in ASME BPVC Section II, Part D, Subpart 1, Tables 2A and 2B.

S_{mc} = allowable stress intensity S_{mc} is 1.1 times the S listed in ASME BPVC Section II, Part D, Subpart 1, Tables 1A and 1B, except S_{mc} is not to exceed 90% of the material's yield strength at temperature shown in ASME BPVC Section II, Part D, Subpart 1, Tables Y-1.

S_f = 85% of the general primary membrane allowable permitted in Mandatory Appendix XXVII, ASME B&PV Code Section III. In the application of Appendix XXVII, S_m, if applicable, is as specified in NE-3112.4(a)(1).

*1: Acceptance Criteria for other than Post-flooding Condition is defined by ASME BPVC, Subsection NE Subparagraphs NE-3221.1 through 3221.4. For Post-flooding Condition, Service Level C limits apply to primary stress, and Service Level B limits apply to primary plus secondary stress, per item 5 of SRP Acceptance Criteria in NUREG-0800, SRP 3.8.2.

*2: Values shown are for a rectangular section. See ASME BPVC, Subsection NE, Sub subparagraph NE-3221.3(d) for other than a solid rectangular section.

*3: N/A = Not applicable. No evaluation required.

*4: The larger of the two values listed is chosen as a limit load

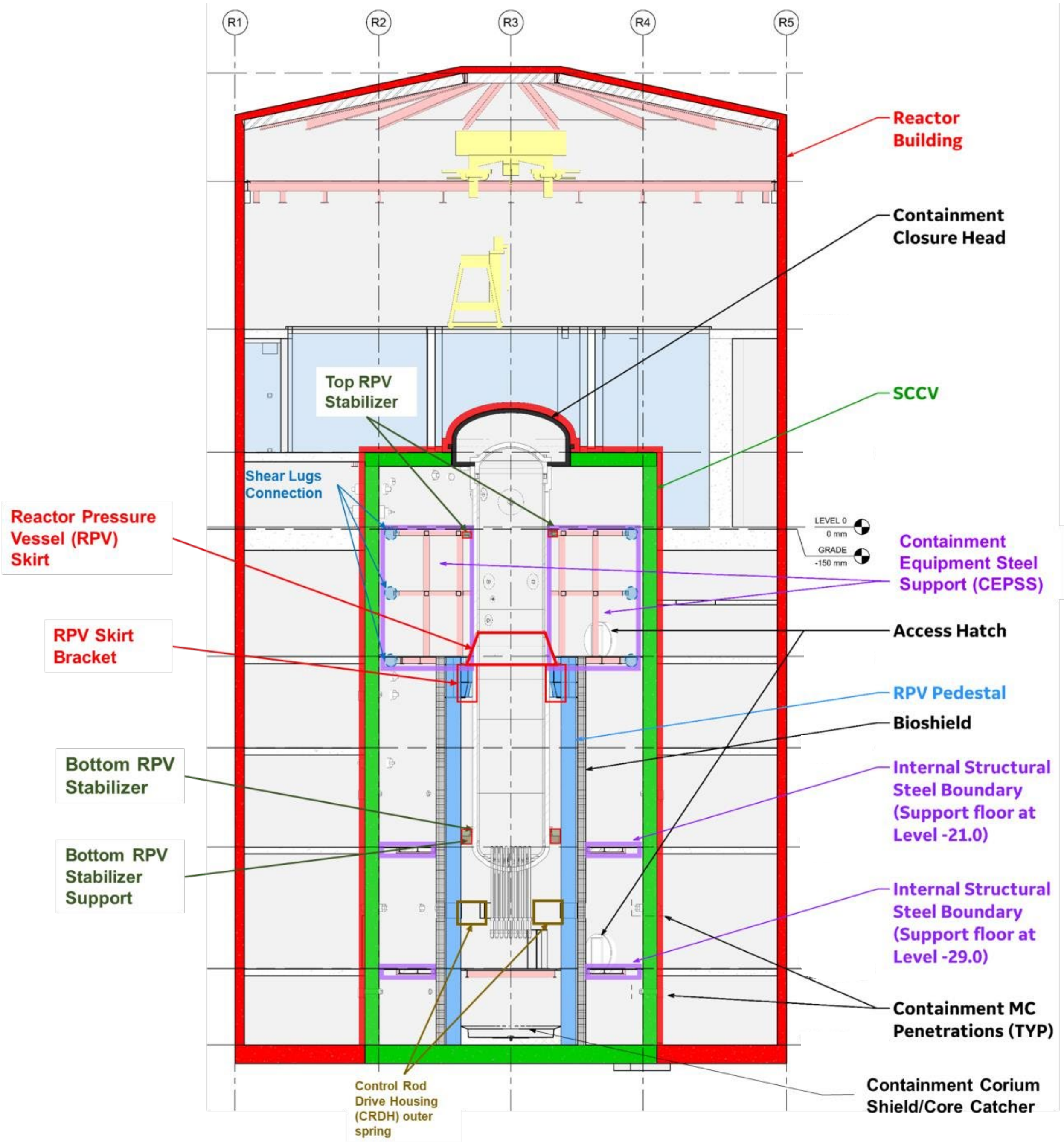


Figure 9B-1: BWRX-300 Integrated RB Structure Including Common Mat Foundation

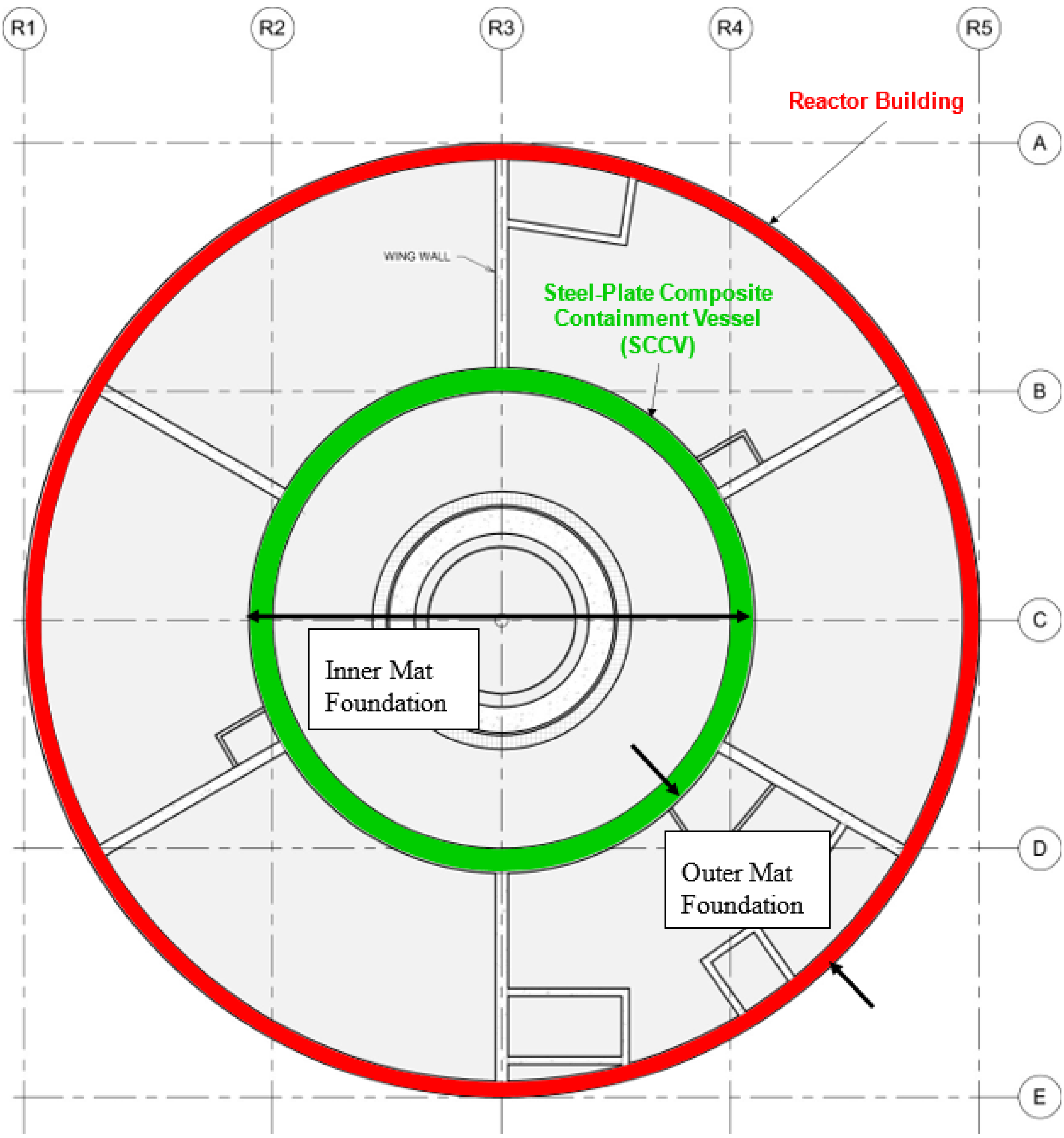


Figure 9B-2: Plan View of BWRX-300 Common Mat Foundation

Notes:

1. RB outer diameter is approximately 36.4 meters.
2. SCCV outer diameter is approximately 19.3 meters.
3. Foundation and embedment depths are provided in Figure 9B-1.

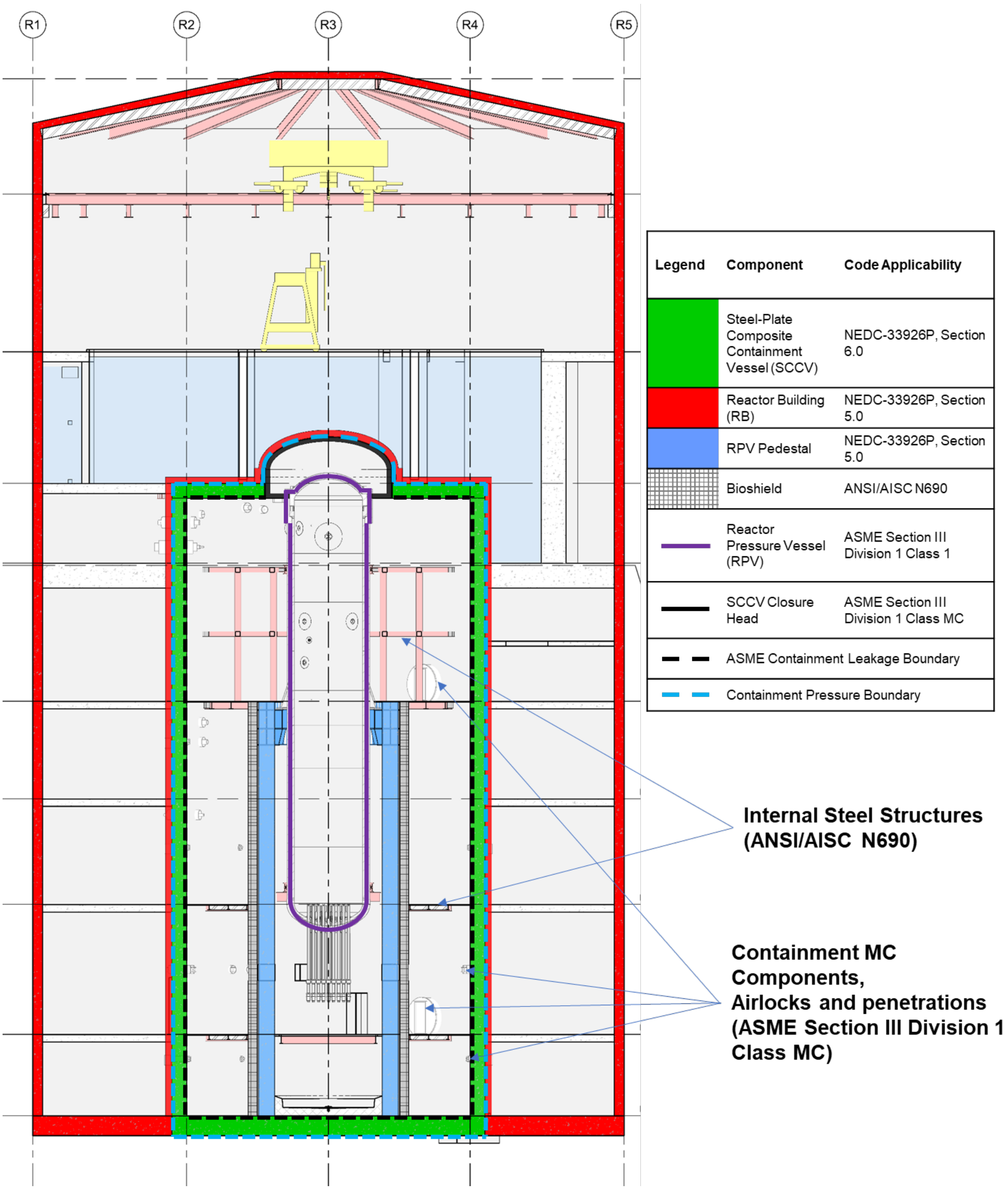


Figure 9B-3: Structural Boundary of the BWRX-300 Containment, Containment Internal Structures and Reactor Building

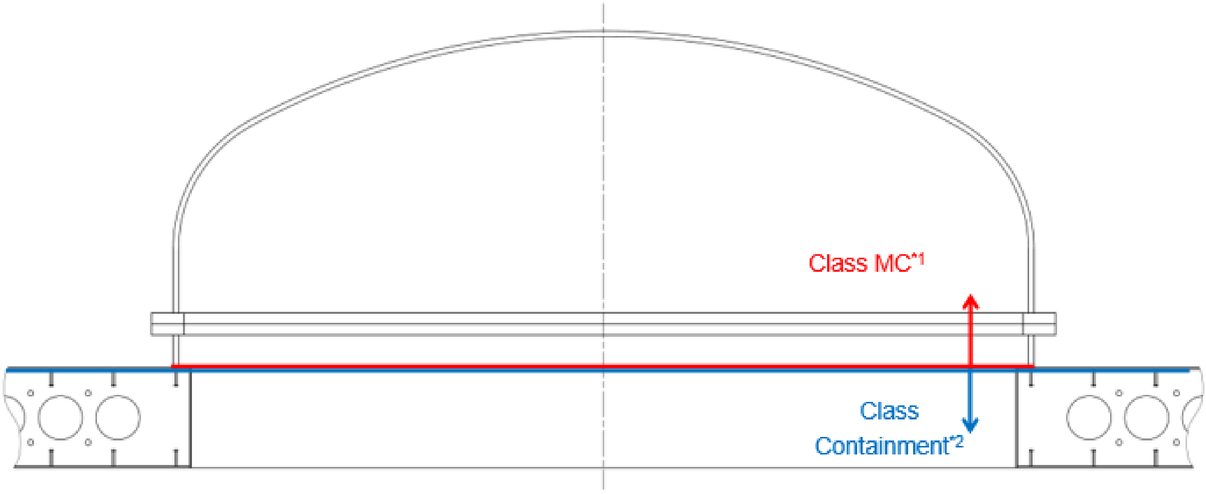


Figure 9B-4: Containment Closure Head Structure Boundary

Notes:

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

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

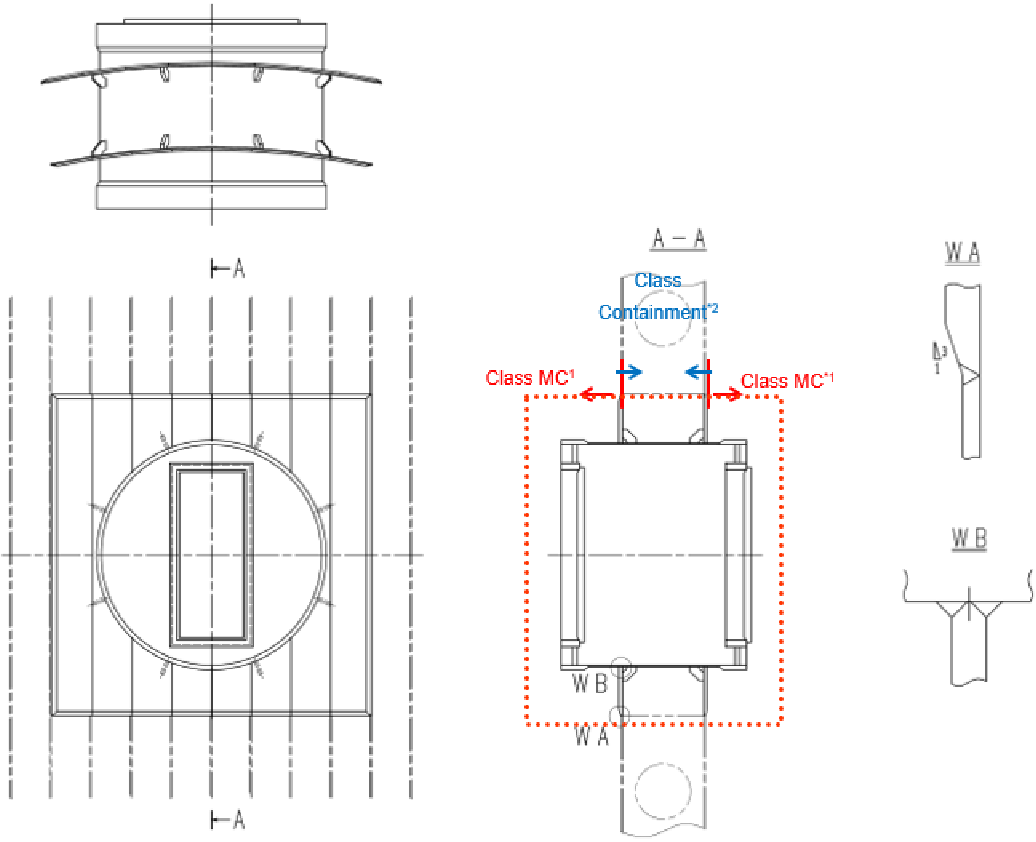


Figure 9B-5: Containment Airlock Code Jurisdictional Boundary

Notes:

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

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

*3: Structural overview is provided for illustration purposes only and will be updated as the design progresses.

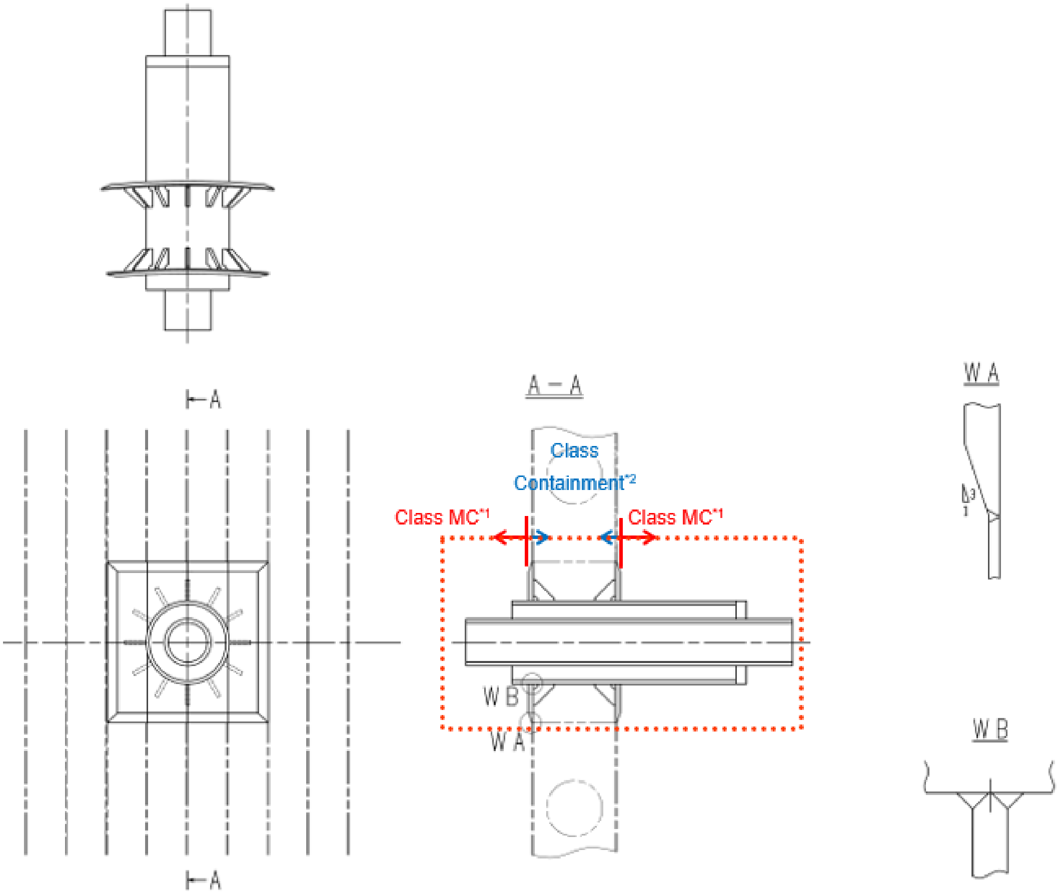


Figure 9B-6: Penetrations Jurisdictional Boundary

Notes:

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

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

*3: Structural overview is provided for illustration purposes only and will be updated as the design progresses.

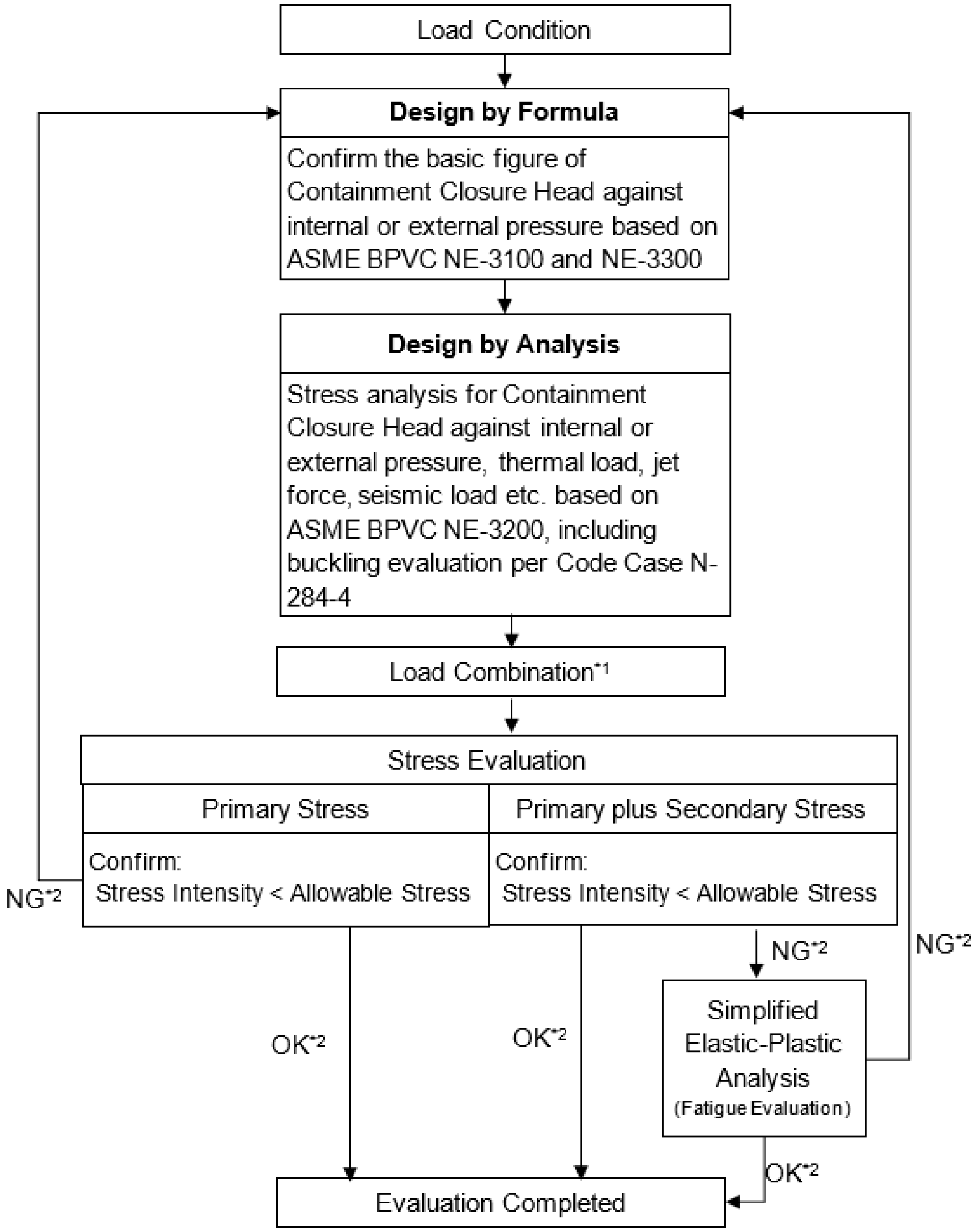


Figure 9B-7: Design Procedures for the Containment Closure Head

Notes:

*1: NUREG-0800, SRP 3.8.2 and USNRC RG 1.57

*2: Ok refers to “Okay” and NG refers to “No Good” in the flowchart

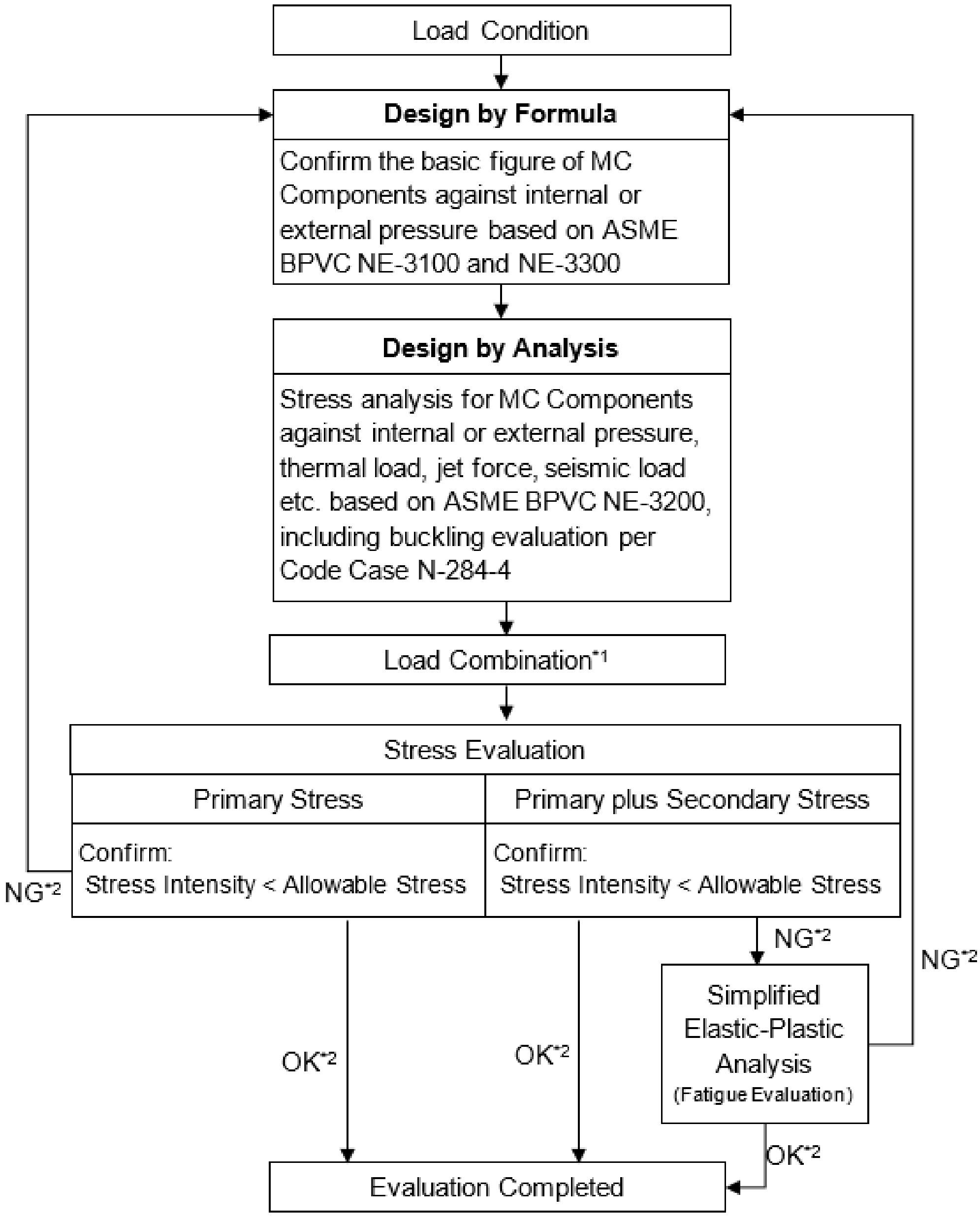


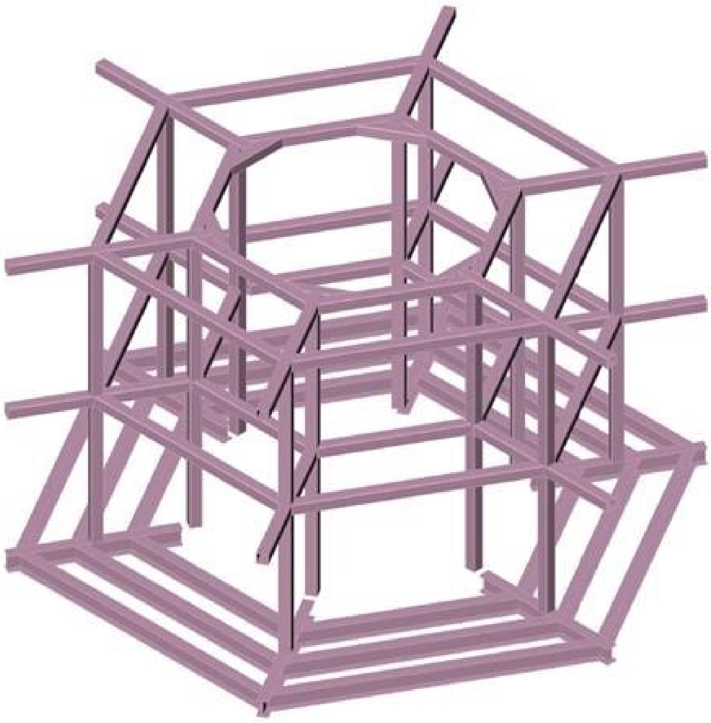
Figure 9B-8: Design Procedures for the Containment Class MC Components

Notes:

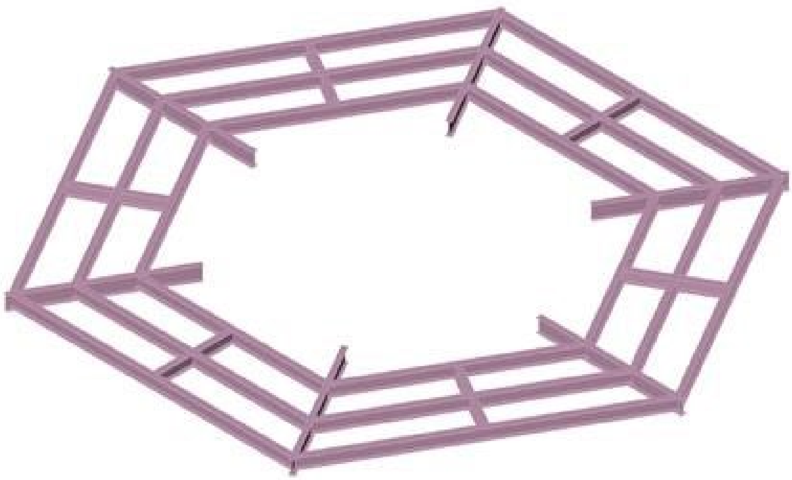
*1: NUREG-0800, SRP 3.8.2 and USNRC RG 1.57

For metal components backed by concrete, refer to NEDC-33926P

*2: OK refers to “Okay” and NG refers to “No Good” in the flowchart



(a) CEPSS



(b) Support floor at Level -21 and -29m

Figure 9B-9: BWRX-300 Containment Internal Structural Steel

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APPENDIX A: CLAIMS, ARGUMENT EVIDENCE ROUTE MAP

A.1 Claims, Argument, Evidence

The Office for Nuclear Regulation (ONR) "SAPs 2014," (Reference 9B-85) 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 Generic Design Assessment (GDA) CAE structure is defined within NEDC-34140P, "BWRX-300 GDA Safety Case Development Strategy (SCDS)," (Reference 9B-86) and is a logical breakdown of an 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 compliance against are identified within the SCDS (Reference 9B-86) and are as follows:

2.1.2: The design of the system/structure has been substantiated to achieve the safety functions in all relevant operating modes.

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 Operating Experience 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: Ageing 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

2.1.6: The BWRX-300 will be designed so that it can be decommissioned safely, using current available technologies, and with minimal impact on the environment and people

2.4.1: Relevant Good Practice (RGP) has been taken into account across all disciplines

2.4.2: Operational Experience (OPEX) and Learning from Experience (LfE) has been taken into account across all disciplines

2.4.3: Optioneering (all reasonably practicable measures 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 Table A-1 below).

It is not the intention to generate a comprehensive suite of evidence to support the derived arguments, as this is beyond the scope of GDA Step 2. 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 2-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:

1. The chapter-specific arguments derived may be supported by existing and future planned evidence sources covering the following topics:
    a. RGP has demonstrably been followed
    b. OPEXs has been taken into account within the design process
    c. All reasonably practicable options to reduce risk have been incorporated within the design.
2. It supports its applicable level 3 sub-claims, defined within the SCDS (Reference 9B-86)

Probabilistic safety aspects of the ALARP argument are addressed within PSR Ch. 15.

In terms of civil structure design aspects that support a future ALARP argument, the BWRX-300 has been designed with modularisation and constructability in mind. The system layouts have been simplified with fewer safety systems and pools of water. RGP has been adopted by locating many of the critical safety systems, including the RPV, below ground level in the Reactor Building. This provides additional protection again external hazards. BWRX-300 Seismic Category 1A structures are also reduced to a minimum, the only one being the RB. The design has also followed OPEX from previous BWRs, particularly, regarding building interaction and consequential risks from system failures. RGP has been used by minimizing the amount of excavation, concrete and the need for backfill materials. Advanced construction methods are used wherever possible with a significant amount of learning taken from the Advanced Boiling Water Reactor (ABWR) modular construction techniques.

Table A-1: Claims, Arguments, Evidence Route Map

L3 No.	Level 3 Chapter Claim:	Chapter 9B Arguments:	Sub-sections and/or reports that evidence the arguments:
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)			
2.1.2	The design of the system/structure has been substantiated to achieve the safety functions in all relevant operating modes.	Safety functions associated with the relevant SSCs have been substantiated during normal operating conditions (including design codes and standards compliance)	9B.1.1.1.2 Safety Design Bases (Common mat foundation) 9B.1.1.3 Structural analysis and design basis 9B.1.1.4 Materials 9B.1.1.11 Performance and safety evaluation 9B.2.1.1.2 Safety Design Bases (containment) 9B.2.1.3 Structural analysis and design basis 9B.2.1.4 Materials 9B.2.1.11 Performance and safety evaluation 9B.2.2.1.2 Safety Design Bases (containment internal structure) 9B.2.2.3 Structural analysis and design basis 9B.2.2.4 Materials 9B.2.2.11 Performance and safety evaluation 9B.2.3.1.2 Safety Design Bases (RB Outside containment) 9B.2.3.3 Structural analysis and design basis 9B.2.3.4 Materials 9B.2.3.11 Performance and safety evaluation 9B.2.4.3 Structural analysis and design basis (RB Pools and liners) 9B.2.4.4 Materials 9B.2.4.11 Performance and Safety Evaluation 9B.3.1.3 Structural analysis and design basis (RWB) 9B.3.1.4 Materials 9B.3.1.11 Performance and safety evaluation 9B.3.2.3 Structural analysis and design basis (CB) 9B.3.2.4 Materials 9B.3.2.11 Performance and safety evaluation 9B.3.3.3 Structural analysis and design basis (TB) 9B.3.3.4 Materials

L3 No.	Level 3 Chapter Claim:	Chapter 9B Arguments:	Sub-sections and/or reports that evidence the arguments:
			9B.3.3.11 Performance and safety evaluation 9B.3.4.3 Structural analysis and design basis (Service Building, Reactor Auxiliary Structures and Fire Water storage Tank and Pump Enclosure) 9B.3.4.4 Materials 9B.3.4.11 Performance and safety evaluation
		A record of safe BWR plant operation and continuous improvement demonstrates a well-founded design	This argument will be addressed at 2-Step GDA
		Safety functions associated with the relevant SSCs have been substantiated during hazard and fault conditions	Safety function will be identified in Chapters 3 & 15 9B.1.1.1.2 Safety Design Bases (common mat foundation) 9B.1.1.11 Performance and safety evaluations 9B.2.1.1.2 Safety Design Bases (containment) 9B.2.1.11 Performance and safety evaluations 9B.2.2.1.2 Safety Design Bases (containment internal structure) 9B.2.2.11 Performance and safety evaluations 9B.2.3.1.2 Safety Design Bases (RB Outside containment) 9B.2.3.11 Performance and safety evaluations 9B.2.4.3 Structural analysis and design basis (RB Pools and liners) 9B.2.4.11 Performance and Safety Evaluation 9B.2.6 Robustness Design of BWRX-300 Seismic Category 1A Structures 9B.3.1.3 Structural analysis and design basis (RWB) 9B.3.1.11 Performance and safety evaluation 9B.3.2.3 Structural analysis and design basis (CB) 9B.3.2.11 Performance and safety evaluation 9B.3.3.3 Structural analysis and design basis (TB) 9B.3.3.11 Performance and safety evaluation 9B.3.4.3 Structural analysis and design basis (Service Building, Reactor Auxiliary Structures and Fire Water storage Tank and Pump Enclosure

L3 No.	Level 3 Chapter Claim:	Chapter 9B Arguments:	Sub-sections and/or reports that evidence the arguments:
			9B.3.4.11 Performance and safety evaluation
		Any shortfalls in safety function substantiation have been identified and assessed to identify any reasonably practicable means to reduce risk	This argument is out of the scope of a 2-Step GDA and will be addressed during a site-specific stage (when evidence is developed)
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 Operating Experience to support reducing risks ALARP	Design evolutions to SSCs have been considered taking into account relevant BWR OPEX, and any reasonably practicable changes to reduce risk have been implemented	NEDC-34137P, "BWRX-300 Design Evolution," (Reference 9B-87)
		The SSCs have been designed in accordance with relevant codes and standards (RGP)	NEDC-34139P, "BWRX-300 UK GDA Codes and Standards Report" (tranche 2 version) plus its associated spreadsheet (Reference 9B-88) This PSR chapter also discusses the codes and standards to which it has been designed
		The SSCs have been designed in accordance with an appropriate suite of design safety principles	The GEH Safety and Design Principles are documented in 006N5064, "BWRX-300 Safety Strategy," (Reference 9B-89) supplemented by 005N9751, "BWRX-300 General Description," (Reference 9B-90). These principles are also presented within PSR Ch. 3 1.
2.1.4	System/structure performance will be validated by suitable testing throughout manufacturing, construction, and commissioning.	SSCs pre-commissioning tests (e.g., NDT) validate the relevant performance requirements	9B.1.1.9 Monitoring, Inspection, Testing and Maintenance (common mat foundation) 9B.2.1.9 Monitoring, Inspection, Testing and Maintenance (containment) 9B.2.2.9 Monitoring, Inspection, Testing and Maintenance (containment internal structure) 9B.2.3.9 Monitoring, Inspection, Testing and Maintenance (RB Outside containment) 9B.2.4.9 Monitoring, Testing, Inspection and Maintenance (RB Pools and liners) 9B.3.1.9 Monitoring, Testing, Inspection and Maintenance (RWB)

L3 No.	Level 3 Chapter Claim:	Chapter 9B Arguments:	Sub-sections and/or reports that evidence the arguments:
			9B.3.2.9 Monitoring, Testing, Inspection and Maintenance (CB) 9B.3.3.9 Monitoring, Testing, Inspection and Maintenance (TB) 9B.3.4.9 Monitoring, Testing, Inspection and Maintenance (Service Building, Reactor Auxiliary Structures and Fire Water storage Tank and Pump Enclosure)
		SSCs commissioning tests (e.g., system level pressure and leak tests) validate the relevant performance requirements	9B.1.1.9 Monitoring, Inspection, Testing and Maintenance (common mat foundation) 9B.2.1.9 Monitoring, Inspection, Testing and Maintenance (containment) 9B.2.2.9 Monitoring, Inspection, Testing and Maintenance (containment internal structure) 9B.2.3.9 Monitoring, Inspection, Testing and Maintenance (RB Outside containment) 9B.2.4.9 Monitoring, Testing, Inspection and Maintenance (RB Pools and liners) 9B.3.1.9 Monitoring, Testing, Inspection and Maintenance (RWB) 9B.3.2.9 Monitoring, Testing, Inspection and Maintenance (CB) 9B.3.3.9 Monitoring, Testing, Inspection and Maintenance (TB) 9B.3.4.9 Monitoring, Testing, Inspection and Maintenance (Service Building, Reactor Auxiliary Structures and Fire Water storage Tank and Pump Enclosure)
		SSCs are manufactured, constructed, and commissioned in accordance with QA arrangements appropriate to their safety classification	PSR Ch. 3 defines this approach. 9B.1.1.4 Materials (common mat foundation). Refers to 9B.2.1.4 (Quality control 9B.2.1.4.7) & 9B.2.3.4 (Quality control 9B.2.3.4.8). 9B.2.1.4.7 Containment Quality Control (RB Containment) 9B.2.2.4 Materials (containment internal structure. Also refers to 9B.2.3.4.8 for RPV pedestal. 9B.2.3.4.8 Quality Control (RB Outside containment) 9B.2.4.4 Materials (RB Pools and liners) 9B.3.1.4 Materials (RWB)

L3 No.	Level 3 Chapter Claim:	Chapter 9B Arguments:	Sub-sections and/or reports that evidence the arguments:
2.1.5	Ageing 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	SSC ageing and degradation mechanisms will be identified during SSC design. These will be assessed to determine how they could potentially lead to SSC failure	9B.2.1.3.3.5 Corrosion Prevention (Containment)
		Appropriate Examination, Maintenance, Inspection and Testing (EMIT) arrangements will be specified taking into account SSC ageing and degradation mechanisms	9B.1.1.9 Monitoring, Inspection, Testing and Maintenance (mat foundation) 9B.2.1.9 Monitoring, Inspection, Testing and Maintenance (containment) 9B.2.2.9 Monitoring, Inspection, Testing and Maintenance (containment internal structure) 9B.2.3.9 Monitoring, Inspection, Testing and Maintenance (RB Outside containment)
		The SSCs that cannot be replaced have been shown to have adequate life	9B.1.1.4 Materials (common mat foundation) 9B.1.1.9 Monitoring, Inspection, Testing and Maintenance 9B.2.1.4 Materials (containment) 9B.2.1.9 Monitoring, Inspection, Testing and Maintenance 9B.2.2.4 Materials (containment internal structure) 9B.2.2.9 Monitoring, Inspection, Testing and Maintenance 9B.2.3.4 Materials (RB Outside containment) 9B.2.3.9 Monitoring, Inspection, Testing and Maintenance 9B.2.4.4 Materials (RB Pools and liners) 9B.2.4.9 Monitoring, Testing, Inspection and Maintenance 9B.3.1.4 Materials (RWB) 9B.3.1.9 Monitoring, Testing, Inspection and Maintenance 9B.3.2.4 Materials (CB) 9B.3.2.9 Monitoring, Testing, Inspection and Maintenance 9B.3.3.4 Materials (TB) 9B.3.3.9 Monitoring, Testing, Inspection and Maintenance 9B.3.4.4 Materials (Service Building, Reactor Auxiliary Structures and Fire Water storage Tank and Pump Enclosure)

L3 No.	Level 3 Chapter Claim:	Chapter 9B Arguments:	Sub-sections and/or reports that evidence the arguments:
			9B.3.4.9 Monitoring, Testing, Inspection and Maintenance
		Ageing and degradation OPEX will be considered as part of the design stage component/materials selection process in order to mitigate SSC failure risk	9B.1.1.4 Materials (common mat foundation) 9B.1.1.9 Monitoring, Inspection, Testing and Maintenance 9B.2.1.4 Materials (containment) 9B.2.1.9 Monitoring, Inspection, Testing and Maintenance 9B.2.2.4 Materials (containment internal structure) 9B.2.2.9 Monitoring, Inspection, Testing and Maintenance 9B.2.3.4 Materials (RB Outside containment) 9B.2.3.9 Monitoring, Inspection, Testing and Maintenance 9B.2.4.4 Materials (RB Pools and liners) 9B.2.4.9 Monitoring, Testing, Inspection and Maintenance 9B.3.1.4 Materials (RWB) 9B.3.1.9 Monitoring, Testing, Inspection and Maintenance 9B.3.2.4 Materials (CB) 9B.3.2.9 Monitoring, Testing, Inspection and Maintenance 9B.3.3.4 Materials (TB) 9B.3.3.9 Monitoring, Testing, Inspection and Maintenance 9B.3.4.4 Materials (Service Building, Reactor Auxiliary Structures and Fire Water storage Tank and Pump Enclosure) 9B.3.4.9 Monitoring, Testing, Inspection and Maintenance
2.1.6	The BWRX-300 will be designed so that it can be decommissioned safely, using current available technologies, and with minimal impact on the environment and people	SSCs decommissioning is considered at the design stage to ensure that safe decommissioning may take place	BWRX-300 Decommissioning planning (PSR Ch. 21). OPEX demonstrates that decommissioning of reactor facilities is facilitated if considered during the design phase: Materials are selected to minimize the quantities of radioactive waste and assisting decontamination. Plant layout is designed to facilitate access for decommissioning or dismantling activities. Future potential requirements for storage of radioactive waste.

L3 No.	Level 3 Chapter Claim:	Chapter 9B Arguments:	Sub-sections and/or reports that evidence the arguments:
		SSCs are designed in order to minimise effects on people and the environment during decommissioning	BWRX-300 Decommissioning planning (PSR Ch. 21). OPEX demonstrates that decommissioning of reactor facilities is facilitated if considered during the design phase: Materials are selected to minimize the quantities of radioactive waste and assisting decontamination. Future potential requirements for storage of radioactive waste. Design of facility is not overly pessimistic leading to larger structures which are more difficult to dismantle and generate more waste
2.4 Safety risks have been reduced as low as reasonably practicable			
2.4.1	Relevant Good Practice (RGP) has been taken into account across all disciplines	Relevant SSCs codes and standards (RGP) are identified	NEDC-34139P (Codes and Standards report (tranche 2 version) plus its associated spreadsheet)
		SSCs have been designed in accordance with relevant codes and standards (RGP)	This PSR chapter also discusses the codes and standards to which it has been designed
		Any shortfalls in codes and standards compliance are identified and assessed to reduce risks ALARP	NEDC-34139P (Codes and Standards report (tranche 2 version) plus its associated spreadsheet) This argument is out of two step GDA scope and will be addressed during site-specific stage.
2.4.2	OPEX and Learning from Experience (LfE) has been taken into account across all disciplines	Design improvements to SSCs have been identified considering relevant OPEX and LfE	NEDC-34137P (Reference 9B-87)
		Any reasonably practicable design changes to reduce risk have been implemented	This argument is out of two step GDA scope and will be addressed during site-specific stage.
2.4.3	Optioneering (all reasonably practicable measures have been	Design optioneering has been performed in accordance with an approved process	006N3139, “BWRX-300 Design Plan,” (Reference 9B-91).

L3 No.	Level 3 Chapter Claim:	Chapter 9B Arguments:	Sub-sections and/or reports that evidence the arguments:
	implemented to reduce risk)	Design optioneering has considered all reasonably practicable measures	006N3139 (Reference 9B-91) NEDC-34137P (Reference 9B-87)
		Any reasonably practicable design changes to reduce risk have been implemented	NEDC-34137P (Reference 9B-87)

APPENDIX B: FORWARD ACTION PLAN

This appendix presents the Forward Action Plan (FAP) for the BWRX-300 UK GDA relating to Civil Structures.

The FAP captures the actions required for the program to progress from GDA to a site-specific phase and captures any commitments made in response to Regulatory Queries and Regulatory Observations.

Table B-1: Forward Action Plan

Action ID	Source	Finding	Forward Actions	Lead Discipline	Delivery Phase
PSR 9B-312	PSR Ch. 9B	Safety Functional Requirements (SFRs) that apply to individual civil engineering SSCs, based on fault and hazard analysis and a decomposition of requirements down from Fundamental Safety Functions (FSFs) are not available.	In GDA Step 2, develop a process for development of SFR schedules and an example SFR schedule for the Reactor Building. The example schedule will be based on available information only, such as US Hazard Identification and Characterization, building specifications and other information that has been used to develop the standard design. It is noted that further development of SFR schedules will be required as additional safety analysis inputs become available post-Step 2.	Civil Engineering	Quarter 2 2025
PSR 9B-279	PSR PSR Ch. 9B	Content relating to aircraft crash is missing: (non-exhaustive): Flow diagram for aircraft impact case including FRS, fire, local scabbing, global overturning, and links to safety studies for other non-civils SSCs.	As part of the specification for the UK Safety Case Manual develop a flow diagram for aircraft impact analysis that identifies the specific procedures needed at each step of the analysis of the sequence progression.	Civil Engineering (Linked to PSR15-3)	Quarter 2 2025
			Deterministic accidental aircraft crash analysis is currently underway within GEH. This will be reviewed and assessed against UK expectations in support of the future Pre-Construction Safety Report.	Civil Engineering (Linked to PSR15.8-147)	Pre-Construction Safety Report