ENCLOSURE 1

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White Paper

NEDO-33988, Revision 0,
BWRX-300 Steel-Plate Composite (SC) Containment Vessel (SCCV) and Reactor Building Structural Design

Non-Proprietary Information

GE Hitachi Nuclear Energy

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BWRX-300 Steel-Plate Composite (SC) Containment Vessel (SCCV) and Reactor Building Structural Design

IMPORTANT NOTICE REGARDING CONTENTS OF THIS REPORT

Please Read Carefully

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REVISION SUMMARY

Revision Number	Description of Change
0	Initial Issue

Acronyms and Abbreviations

Term	Definition
ABWR	Advanced Boiling Water Reactor
ACI	American Concrete Institute
ACT	Advanced Construction Technology
AISC	American Institute of Steel Construction
ANSI	American National Standards Institute
ASCE	American Society of Civil Engineers
ASME	American Society of Mechanical Engineers
B&PV	Boiler and Pressure Vessel
BWR	Boiling Water Reactor
CNSC	Canadian Nuclear Safety Commission
CPA	Construction Permit Application
CSA	Canadian Standards Association
DBA	Design Basis Accident
DBE	Design Basis Earthquake
DEC	Design Extension Condition
DG	Design Guide
ESBWR	Economic Simplified Boiling Water Reactor
GEH	GE Hitachi Nuclear Energy Americas LLC
HGNE	Hitachi-GE Nuclear Energy Ltd
ILRT	Integrated Leak Rate Test
LTC	Licence to Construct
LTR	Licensing Topical Report
NRC	Nuclear Regulatory Commission
NRIC	National Reactor Innovation Center
OBE	Operating Basis Earthquake

Term	Definition
PGA	Peak Ground Acceleration
PIE	Postulated Initiating Event
QC	Quality Control
RB	Reactor Building
RG	Regulatory Guide
RPV	Reactor Pressure Vessel
SC	Steel-plate composite
SCCV	Steel-Plate Composite Containment Vessel
SEI	Structural Engineering Institute
SIT	Structural Integrity Test
SMR	Small Modular Reactor
SSC	Structure, System, and Component
SSI	Soil-Structure Interaction
SRP	Standard Review Plan
SSE	Safe Shutdown Earthquake
ZPA	Zero Period Acceleration

1.0 INTRODUCTION

1.1 Purpose

The purpose of this white paper is to request feedback from the Canadian Nuclear Safety Commission (CNSC) and Nuclear Regulatory Commission (NRC) regarding the proposed use of steel-plate composite (SC) materials for the GE Hitachi Nuclear Energy Americas LLC (GEH) BWRX-300 Steel-Plate Composite Containment Vessel (SCCV) and Reactor Building (RB) structures.

1.2 Scope

The scope of this document includes the following:

- A general description of the BWRX-300 is provided, including description of the integrated RB consisting of the Containment structure, including SCCV and containment internal structures, and the RB structure, and including a general overview of SC materials including Steel Bricks™ (References 5-1 and 5-2).
- A technical evaluation of the proposed use of Steel Bricks™ for the integrated RB is described, including description of the Steel Bricks™ materials and requirements for design, analysis, fabrication, construction, examination, testing, marking, stamping and preparation of reports.
- A technical evaluation of the proposed codes and standards applicable to the proposed use of Steel Bricks™ for the integrated RB is described, including development of requirements for the design, analysis, qualification, and testing of Steel Bricks™ based on the results of the National Reactor Innovation Center (NRIC) Demonstration Program.
- A regulatory evaluation used to address both CNSC and NRC regulatory requirements and guidance in future licensing applications is described, including proposed content for a future Licensing Topical Report (LTR).

The metal components not backed by concrete at the structural boundary of the Containment structure are not in the scope of this document.

2.0 BWRX-300 GENERAL DESCRIPTION

BWRX-300 is an approximately 300 MWe, water-cooled, natural circulation Small Modular Reactor (SMR) utilizing simple passive safety systems driven by natural phenomena. It is developed by GEH in the U.S. and Hitachi-GE Nuclear Energy Ltd (HGNE) in Japan. It is the tenth generation of the Boiling Water Reactor (BWR). BWRX-300 is an evolution of the U.S. NRC-certified design for the 1,520 MWe Economic Simplified Boiling Water Reactor (ESBWR).

The BWRX-300 power block consists of several structures as shown in Figure 2-1. Each structure houses components that perform the various functions which result in the generation of electricity in the Turbine Building. The RB structure houses the main function of steam generation and is shown in Figure 2-1 as the circular structure. It is separated from the rest of the surrounding power block structures by seismic gaps, limiting the physical interaction between the RB and power block structures during a seismic event.

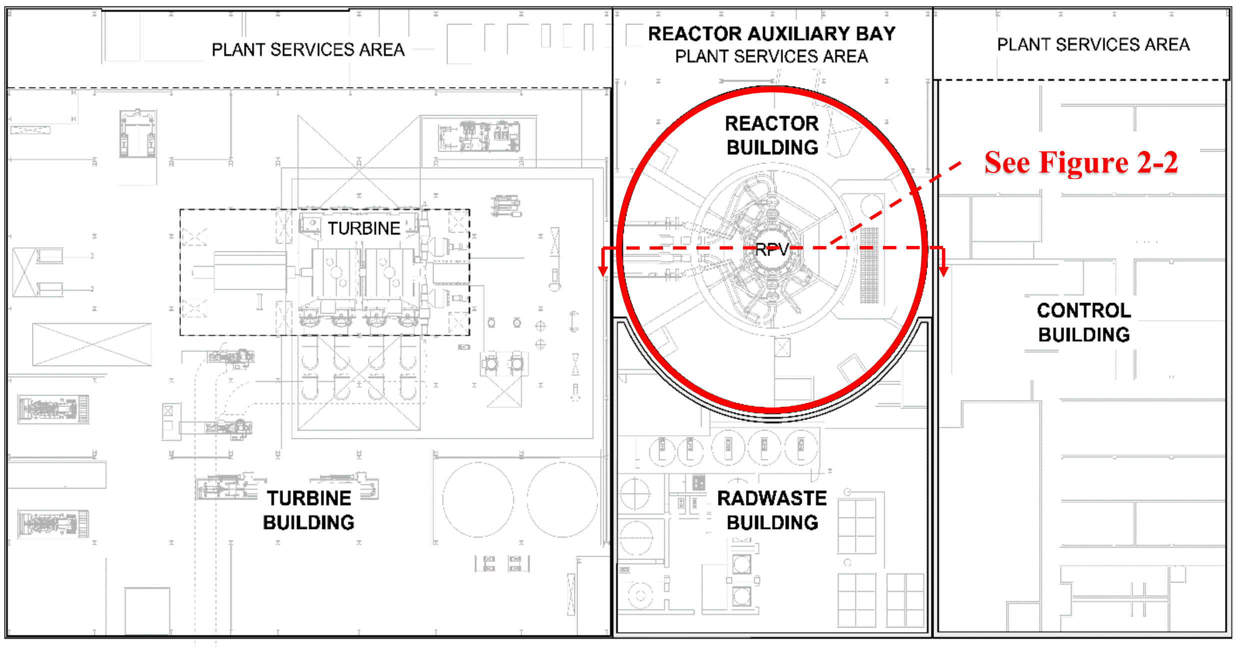

Figure 2-1: BWRX-300 Power Block Plan View

2.1 Reactor Building and Containment Structures Overview

The BWRX-300 integrated RB including walls, floors, and RB roof act in an integrated manner to provide suitable load path for gravity and lateral loads. The RB structure is a cylindrical-shaped, shear wall building that is deeply embedded to an approximate depth of 36 meters below grade. The walls, floors, roof, and basemat of the RB structure are primarily constructed using Steel Bricks™.

The Containment structure consists of the SCCV which is constructed of Steel Bricks™ and other metal components that are not made of Steel Bricks™, such as the steel containment closure head, hatches, and penetrations. The metal components not backed by concrete at the Containment boundary are American Society of Mechanical Engineers (ASME) Class MC components and are designed and constructed in accordance with ASME, Section III, Division 1, Subsection NE, and are not within the scope of this document.

The Containment structure, including SCCV and containment internal structures, and the RB structure are integrated at the basemat and at the connections of wing walls and elevated floor slabs, including the reactor well slab and the walls. Figures 2-2 and 2-3 show the 3D and orthogonal representation, respectively, of the typical integrated RB cross-section and depict the finished grade level. In these figures, the boundary of the RB is shown in red, and Containment is shown in green.

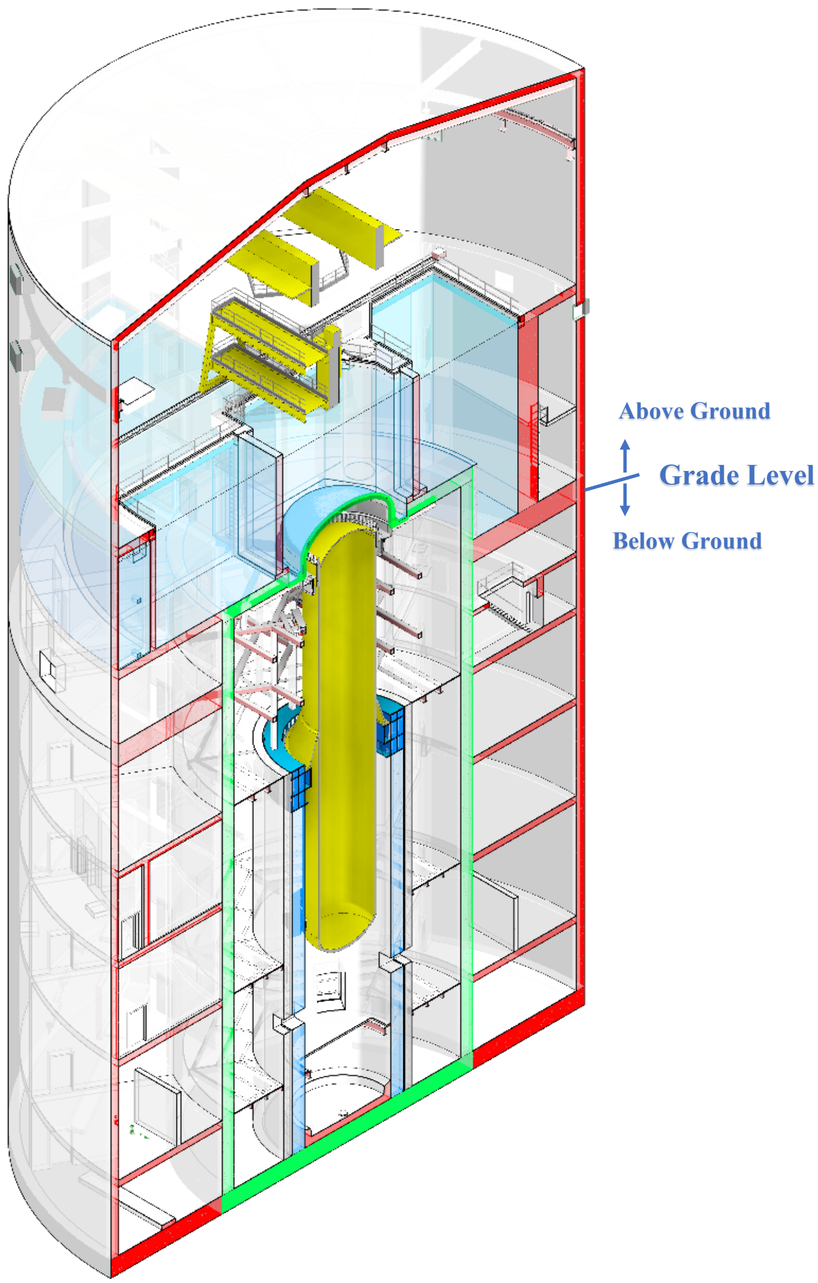

Figure 2-2: 3D Depiction of Integrated Reactor Building

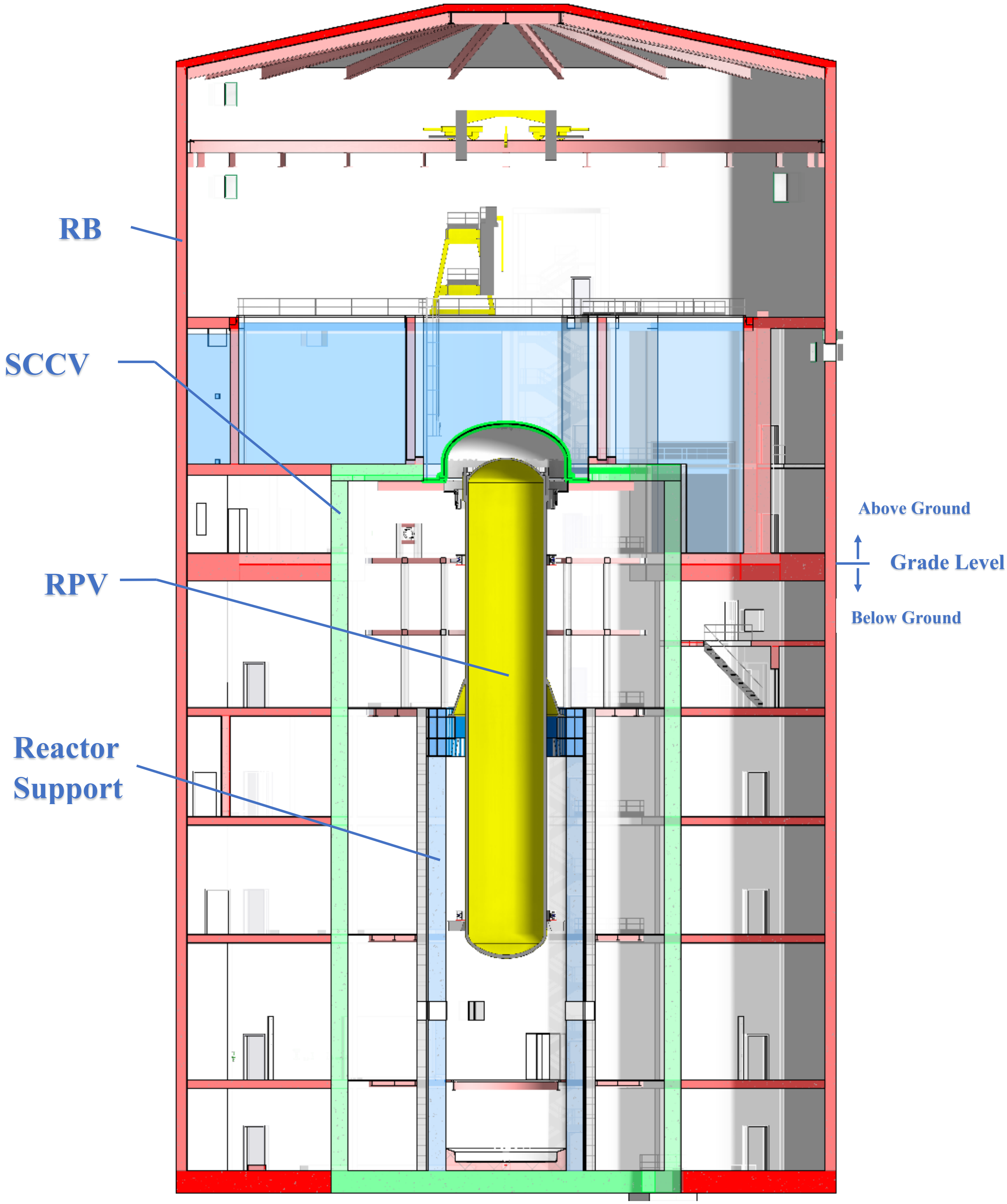

Figure 2-3: Section View of Integrated Reactor Building

The Containment and RB structures house the Reactor Pressure Vessel (RPV), the reactor support and safety systems, and most of the vital and non-vital power supplies and equipment. As evident in Figures 2-2 and 2-3, most of the Containment and RB structures are located below grade, with the above grade portion of RB structure housing the refueling floor, refueling and fuel handling systems, spent fuel pool, and polar crane. The RB roof is made of SC modules supported by structural steel roof beams.

The primary functions of the RB are as follows:

- The RB houses and structurally supports the Containment structure, the RPV, the reactor support structure of the primary reactor system and fuel handling equipment, biological shielding, and associated equipment and structures.
- The RB provides adequate space for the operation, maintenance, and removal of equipment housed within the Containment structure during periodic maintenance.
- The RB provides protection for safety equipment from environmental and natural hazards phenomena, such as floods, winds, tornadoes, and earthquakes.
- The RB provides protection for safety equipment from external hazards, such as explosions and missiles from nearby transportation/industry or aircraft impact.

The Containment structure including the SCCV houses the RPV and the containment internal structures, including the SC internal pedestal that supports the RPV. The BWRX-300 Containment design is based upon GEH BWR experience and fleet performance, including the following features:

- Containment size comparable to a small BWR drywell
- Containment peak accident pressure and temperatures within existing BWR experience base
- Containment load simplified when compared to conventional BWRs with pressure suppression containments, which must account for hydrodynamic loading
- Nitrogen-inerted Containment same as BWR Mark I, BWR Mark II, and Advanced Boiling Water Reactor (ABWR) containments
- Pressure and temperature during normal operation maintained by fan coolers, similar to existing BWRs

2.2 Steel-Plate Composite (SC) Structures

SC structures are proven structural systems with demonstrated structural performance that enable ease of fabrication and construction and have been widely used in commercial and nuclear industry. Previous designs of SC structures consisted of steel faceplates with concrete fill, with steel anchors providing composite behavior between faceplates, and concrete and steel tie bars/rods providing structural integrity and acting as shear reinforcement (see Figure 2 4). The Westinghouse designed and U.S. NRC-certified AP1000® pressurized water reactor uses these traditional designs of SC structural modules.

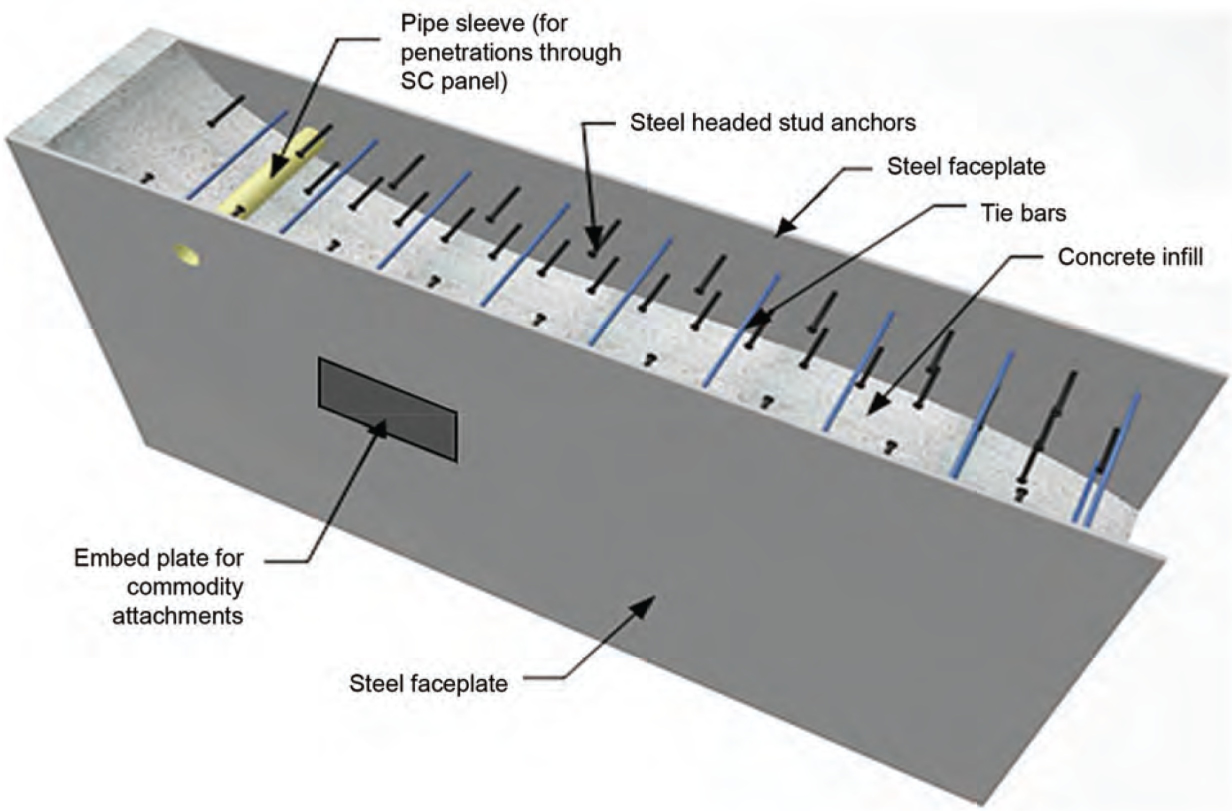

Figure 2-4: Typical Steel Plate Composite Structure (Reference 5-2)

2.3 Steel Bricks™ Overview

Steel Bricks™ is a patented advanced SC modular system used in the construction of most walls and floors of the integrated RB, wherein the use of steel tie bars/rods in traditional SC systems is replaced by steel diaphragms created as part of the fabrication process. The use of Steel Bricks™ improves modularization and quality because it allows fabrication and assembly of modules to occur in a controlled manner remotely to the immediate construction site, minimizing the amount of critical path work needed to complete construction of the integrated RB in the field.

Figure 2-5 shows typical fabrication steps of Steel Bricks™ consisting of individual L-shaped or U-shaped bricks fabricated by:

(i) cutting the steel plate to the desired shape,

(ii) folding the steel plate in a press to create an L-shaped section,

(iii) welding shear studs to the inside surfaces of the plates, if needed by design, and

(iv) assembling the wall or floor modules by welding the individual L-shapes to each other into U-shaped bricks.

Steel Bricks™ eliminates the need for any additional tie bars or systems. Instead of ties, Steel Bricks™ provides stiffer and stronger diaphragm plates to provide stiffness, strength, and stability to the steel modules. Properly sized holes in the diaphragm plates provide a pathway for concrete to flow between U-shaped bricks and provide effective steel area as shear reinforcement.

Figure 2-5: Construction of Steel Bricks™ (Reference 5-1)

3.0 TECHNICAL EVALUATION

Current design codes do not address the use of SC systems as a Containment pressure boundary. Therefore, the design rules for the SCCV are based on proposed alternatives to the ASME Boiler and Pressure Vessel (B&PV) Code, Section III, Rules for Construction of Nuclear Facility Components, Division 2, Code for Concrete Containments, Subsection CC, Concrete Containments, Articles CC-1000 through CC-8000 (Reference 5-3), for materials, design, fabrication, construction, examination, testing, marking, stamping and preparation of reports for the BWRX-300 SCCV, including Division 2 Appendices to the extent they apply to an SC containment without reinforcing steel or tendons.

In addition, the design rules for the RB and the containment internal structures that are not part of the Containment pressure boundary are based on codes and standards with some proposed alternatives to cover design elements beyond the scope of current standards. The proposed alternatives for the RB and the containment internal structures design using Steel Bricks™, as well as the SCCV design using Steel Bricks™, are supplemented by a test program that is being performed under the NRIC Advanced Construction Technology (ACT) project in the United States. This program is known as the BWRX-300 Steel Bricks™ NRIC Demonstration Program and is described in Section 3.2.

3.1 Design Rules

The following describes the codes and standards proposed for use in the design of the BWRX-300 integrated RB, consisting of the Containment structure, including SCCV and containment internal structures, and the RB structure, along with alternative approaches as they apply to Canada and the U.S.

In Canada, nuclear safety-related structure design is governed by Canadian Standards Association (CSA) N291 (Reference 5-4) guidance, with CSA N287 (Reference 5-5) series providing guidance for concrete containment structures. Seismic qualification of structural design is governed by CSA N289.3 (Reference 5-6) guidance.

The CSA N287 series of standards does not include provisions for SC containments. Clause 4.3 of CSA N287.3 permits the use of alternate design methods for design of concrete containments in Canada. Design requirements for the BWRX-300 containment discussed in Subsection 3.1.2 meet the intent and ensure a level of safety and performance commensurate with CSA N287.

3.1.1 Reactor Building

The codes and standards listed below provide the BWRX-300 RB design requirements.

Canadian Codes & Standards

- CSA N289.3 (Reference 5-6):
    - Provides design procedures for seismic qualification of RB.
- CSA N291 (Reference 5-4):
    - Applies to RB excluding Containment.

- Does not preclude use of SC structural systems (including Steel Bricks™), but does not contain specifics for SC.
- Clause 6.1.2 of CSA N291 permits use of alternative design methods.

U.S. Codes & Standards

- American National Standards Institute (ANSI)/American Institute of Steel Construction (AISC) N690-18, Appendix N9 (Reference 5-7):
  - Includes design requirements for SC structural systems (including Steel Bricks™) for nuclear safety-related (non-Containment) structures, with application limited to structural walls.
  - Endorsed by NRC in Regulatory Guide (RG) 1.243 (Reference 5-8).
  - AISC Design Guide (DG) 32 for SC structural systems includes design examples (Reference 5-2).
  - The new LTR (see Section 4.0) provides supplemental design requirements to address the application of SC structures beyond those in current ANSI/AISC N690-18 (Reference 5-7).
  - ANSI/AISC N690-18 includes a provision NA1 which states "For a structural system or construction within the scope of the Nuclear Specification where conditions are not covered by the Nuclear Specification, it is permitted to base the adequacy of the designs on tests, analysis or successful use, subject to the approval of the authority having jurisdiction."
- American Society of Civil Engineers (ASCE)/Structural Engineering Institute (SEI) 43-19 (Reference 5-9):
  - Includes requirements specific to SC structures design (applicable to Steel Bricks™) with focus on:
    - Evaluation of seismic demands,
    - Calculation of structural capacities, and
    - Ductile detailing requirements.
  - Invokes ANSI/AISC N690 (Reference 5-7) for stiffness, capacity, and detailing requirements, and supplements ANSI/AISC N690 by providing damping values and acceptance criteria of SC structures (including Steel Bricks™).

3.1.2 Steel-Plate Composite Containment Vessel (SCCV)

The codes and standards listed below provide the BWRX-300 Containment design requirements.

Canadian Codes & Standards

- CSA N289.3
  - Provides design procedures for seismic qualification of Containment.

U.S. Codes & Standards

- ASME B&PV Code (Reference 5-3):
  - Currently, ASME B&PV Code Section III does not provide requirements for an SCCV (i.e., Containments made from SC structures).
    - GEH has submitted a Code Case to ASME B&PV Code Section III to provide design requirements for an SCCV based on the framework of ASME B&PV Code Section III, Division 2, for Concrete Containments.
      - The design guidelines in the code case for the SCCV are based on the allowable stress design methodology of ASME Section III, Divisions 1 and 2, for Metal Containments and Concrete Containments, respectively, using current research data for SC structures.
      - Reviews by the applicable ASME Section III Code Committees are currently underway, but the approval timeline is not known. Therefore, GEH plans to submit an LTR (see Section 4.0) including the proposed content of the ASME Code Case to request NRC authorization pursuant to 10 CFR 50.55a(z) to use proposed alternatives to the ASME B&PV Code, Section III, Rules for Construction of Nuclear Facility Components, Division 2, Code for Concrete Containments, Subsection CC, Concrete Containments, Articles CC-1000 through CC-8000, for materials, design, fabrication, construction, examination, testing, marking, stamping and preparation of reports for the BWRX-300 SCCV, including Division 2 Appendices to the extent they apply to an SC containment without reinforcing steel or tendons.

3.2 National Reactor Innovation Center (NRIC) Demonstration Program Overview

The main objectives of the NRIC Demonstration Program (NRIC Project) include:

1. Demonstration of the structural performance of Steel Bricks™, and
2. Development and demonstration of the efficient fabrication, installation, and construction processes for use of Steel Bricks™ for nuclear safety-related applications.

The program is comprised of two phases. NRIC Phase 1 (Detailed Design and Structural Performance Testing) has a duration of approximately 12 months with an estimated completion date of year-end 2022. NRIC Phase 2 (Construction, Testing, and Decommissioning Activity), awarded upon successful completion of NRIC Phase 1, has an estimated duration of 2.5 to 3 years with an estimated completion of March 2026. Figure 3-1 shows the schedules for both phases.

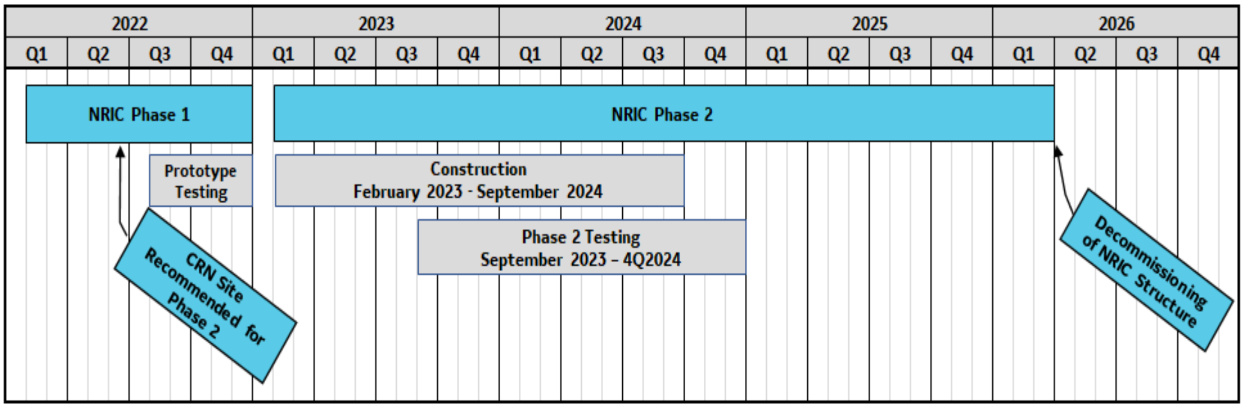

Figure 3-1: NRIC Phase 1 and Phase 2 Estimated Schedule

The objectives of the prototype test (NRIC Phase 1) are to demonstrate the structural performance of Steel Bricks™ under design basis and beyond design basis loading conditions. A total of 15 Steel Bricks™ scaled prototype specimens are constructed and tested for various loading conditions applicable for containment (i.e., pressure-retaining) and non-containment applications. Table 3-1 summarizes the prototype testing for NRIC Phase 1 to evaluate the performance of SC systems under different loading conditions. As such, ANSI/AISC N690, Appendix N9 for SC and the proposed LTR (Section 4.0) can be used, with modifications as applicable, in the design and construction of SC structures made of Steel Bricks™. Numerical simulations are also conducted to benchmark the computation models against the prototype test results. CNSC Staff and NRC Staff have been invited to observe the NRIC Project.

Table 3-1: Brief Summary of Steel Bricks™ NRIC Phase 1 Prototype Tests

Test Type	Prototype	Number of Specimens	Note
Out-Of-Plane Shear	Basemat	2	Specimens representing basemat subjected to out-of-plane flexure and shear loading. Verify that the out-of-plane strength can be calculated conservatively using ANSI/AISC N690 code provisions.
Bi-Axial Tension	SCCV	3	Simulate the bi-axial loading condition induced by accidental pressure and thermal loading conditions. Verify that Steel Bricks™ SCCV can withstand accidental pressure and thermal loading conditions with acceptable steel strain limits.

Test Type	Prototype	Number of Specimens	Note
In-Plane Shear	SCCV-to-Basemat Connection	2	Simulate the design demands induced by Design Basis Earthquake (DBE) and accident thermal loading. Verify that the connection details as well as splicing details are strong enough so that Steel Bricks™ SCCV-to-basemat connections can withstand design demands without connection failures.
In-Plane Shear + Out-Of-Plane Shear	RB-to-Basemat Connection	2	Simulate the design demands induced by DBE and lateral earth pressures. Verify that the connection details are strong enough so that Steel Bricks™ RB-to-basemat connections can withstand design demands without connection failure.
Missile Impact	RB	6	Verify the impact resistance of Steel Bricks™ against missile impact.

In addition to the NRIC Phase 1 effort providing relevant structural performance data for the Steel Bricks™ qualification, NRIC Phase 1 also supports the development of design details for efficient and optimized installation guidelines and procedures for NRIC Phase 2.

NRIC Phase 2 focuses on the actual Demonstration Plant construction, including Quality Control (QC) measures in accordance with certain key criteria. The life cycle for Steel Bricks™ begins with fabrication and continues through transport, assembly, erection, construction, turnover, and decommissioning. During the Steel Bricks™ life cycle, QC inspection data is collected to determine what QC processes produce the optimal quality approach. Steel Bricks™ are fabricated in a shop, assembled at the site into modules, lifted by crane, placed in final position, and connected inside the excavated pit to construct the Demonstration Plant. Different schemes for module assembly, rigging, below the hook devices and lift plans are evaluated for the design to ensure the modular systems do not exceed allowable stress during the erection evolution.

The NRIC Phase 2 Demonstration Plant includes a partial-scale structure, with the installation approaches, techniques, and procedures evaluated during the demonstration mirroring conditions anticipated during eventual full-scale BWRX-300 construction.

4.0 REGULATORY EVALUATION

LTR NEDC-33926P, "BWRX-300 Steel-Plate Composite (SC) Containment Vessel (SCCV) and Reactor Building Structural Design," (Reference 5-10) is planned for submittal to both the CNSC and NRC. The LTR is specific to the design conditions and parameters applicable for the integrated RB design and presents the primary licensing path and regulatory basis for a Construction Permit Application (CPA) in the U.S. or a Licence to Construct (LTC) Application in Canada. The LTR documents and provides justification for the design requirements and regulatory basis for acceptability of use of Steel Bricks™ for the BWRX-300 design. The LTR also includes additional requirements for non-containment nuclear safety-related applications to supplement the requirements for SC (including Steel Bricks™) structures in ANSI/AISC N690.

The results from the prototype tests in NRIC Phase 1 as described in Section 3.2 provides confirmatory data to demonstrate the structural performance of the Steel Bricks™ system for various loading conditions and help inform decisions on the acceptability of the design requirements and acceptance criteria for the integrated RB design in the LTR. Additionally, the Steel Bricks™ fabrication, installation, and construction related design activities during NRIC Phase 1 help to establish the LTR content to ensure compliance with the applicable standards and regulatory requirements. Although NRIC Phase 2 is not directly required for the LTR or for any LTC Application or CPA submittals, any lessons learned during the actual construction of the Demonstration Plant would help improve construction work processes, including examination, inspection, and testing, during future actual construction of the integrated RB.

Figure 4-1 shows the estimated schedule for the LTR development, submittal, and anticipated regulatory reviews.

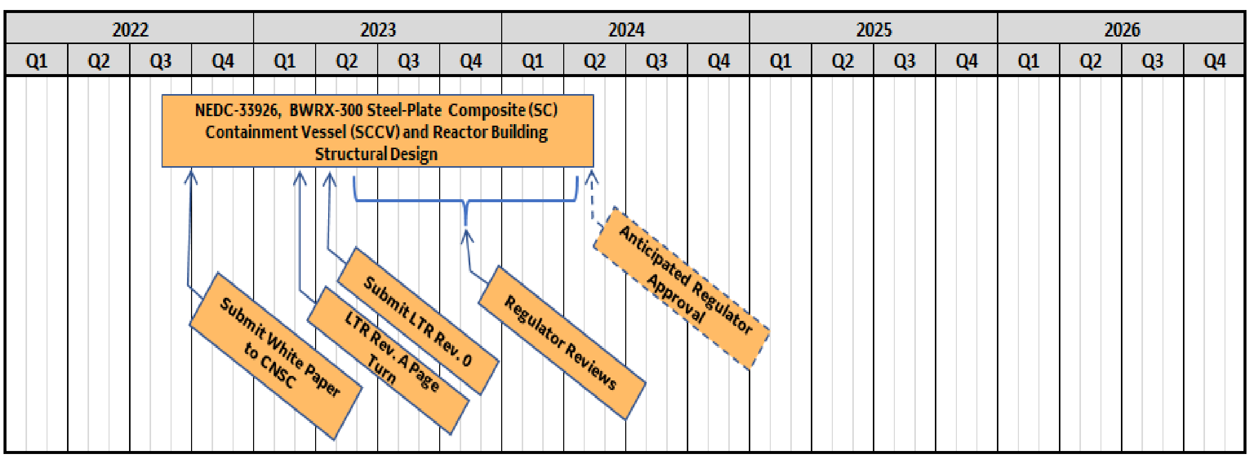

Figure 4-1: LTR Schedule for Submittal and Anticipated Reviews

4.1 Regulatory Requirements and Guidance

The following regulatory requirements and guidance are applicable to the integrated RB and are evaluated in the LTR to either demonstrate compliance or to propose applicable alternative approaches to these regulatory requirements and guidance.

4.1.1 CNSC Regulatory Requirements and Guidance

REGDOC-1.1.2 (Reference 5-11) includes, but is not limited to, the following relevant requirements and guidance:

- Section 4.5.5 describes the requirements for presenting relevant information on the design of the site layout and on civil engineering works and structures associated with the nuclear facility, with sufficient detail for CNSC staff to verify that the design is in accordance with Sections 7.15 and 8.6.2 of REGDOC-2.5.2, Design of Reactor Facilities: Nuclear Power Plants.

REGDOC-2.5.2 (Reference 5-12) includes, but is not limited to, the following relevant requirements and guidance:

- Section 5.4 requires identification of the codes and standards that are used for the plant design, and an evaluation of those codes and standards for applicability, adequacy, and sufficiency to the design of Structures, Systems, and Components (SSCs) important to safety.
- Section 6.6 requires development of a facility layout that considers Postulated Initiating Events (PIEs) to enhance protection of SSCs important to safety with the final design reflecting an assessment of options, demonstrating that an optimized configuration has been sought for the facility layout.
- Section 7.1 requires that the design authority classifies SSCs using a consistent and clearly defined classification method. The SSCs are to be designed, constructed, and maintained such that their quality and reliability is commensurate with this classification.
- Sections 7.4.1 and 7.4.2 require that SSCs important to safety be designed and located in a manner that minimizes the probability and effects of hazards (e.g., fires and explosions) caused by external or internal events, and that all natural and human-induced external hazards that may be linked with significant radiological risk be identified. External hazards which the plant is designed to withstand are to be selected and classified as Design Basis Accidents (DBAs) or Design Extension Conditions (DECs).
- Section 7.7 requires that all pressure retaining SSCs be protected against overpressure conditions, and be classified, designed, fabricated, erected, inspected, and tested in accordance with established standards. Section 7.7 also requires that, for DECs, relief capacity be sufficient to provide reasonable confidence that pressure boundaries credited in severe accident management will not fail.
- Section 7.13.1 requires that the design authority ensures that seismically qualified SSCs important to safety are qualified to a DBE and that they are categorized accordingly.
- Sections 7.15.1, 7.15.2, and 7.15.3 require that the design specifies the required performance for the safety functions of the civil structures in operational states, DBAs and DECs, that it enables implementation of periodic inspection programs for structures important to safety in order to verify that the as-constructed structures meet their functional and performance requirements, and that the lifting and handling of large and heavy loads, particularly those containing radioactive material, be considered in the design.

- Section 7.17 requires that the design takes due account of the effects of aging and wear on SSCs, including additional requirements provided in REGDOC-2.6.3, Aging Management (Reference 5-13).
- Section 7.22 requires that the design provides physical features such as protection against design-basis threats (DBTs), in accordance with the requirements of the Nuclear Security Regulations.
- Section 8.6 requires that each nuclear power reactor be installed within a containment structure, to minimize the release of radioactive materials to the environment during operational states and DBAs. Containment is to also assist in mitigating the consequences of DECs. In particular, the containment and its safety features are to be able to perform their credited functions during DBAs and DECs, including melting of the reactor core. To the extent practicable, these functions shall be available for events more severe than DECs.
- Section 11 allows the use of alternative approaches to the requirements in REGDOC-2.5.2 where:
    1. the alternative approach would result in an equivalent or superior level of safety,
    2. the application of the requirements in this document conflicts with other rules or requirements, and
    3. the application of the requirements in this document would not serve the underlying purpose or is not necessary to achieve the underlying purpose.

4.1.2 NRC Regulatory Requirements

10 CFR 50.34(f), “Additional TMI-related requirements,” requires license applications to provide sufficient information to describe the nature of the studies required, how they are conducted, and a program to ensure that the results of these studies are factored into the final design of the facility, and the studies must be submitted as part of the Final Safety Analysis Report (FSAR). This includes the capability of the containment to resist (1) those loads that are generated by pressure and dead loads during an accident that releases hydrogen generated from 100-percent fuel clad metal-water reaction and accompanied by either hydrogen burning or added pressure from post-accident inerting, and (2) those loads that are generated as a result of an inadvertent full actuation of a post-accident inerting hydrogen control system, excluding seismic or design-basis accident loadings.

10 CFR 50.44, “Combustible gas control for nuclear power reactors,” requires compliance with requirements that include the capability of the containment of new plants to resist those loads associated with combustible gas generation from a metal-water reaction of the fuel cladding. An applicant must perform an analysis that demonstrates containment structural integrity. This demonstration must use an analytical technique that is accepted by the NRC and include sufficient supporting justification to show that the technique describes the containment response to the structural loads involved. The analysis must address an accident that releases hydrogen generated from 100 percent fuel clad-coolant reaction accompanied by hydrogen burning. Systems necessary to ensure containment integrity must also be demonstrated to perform their function under these conditions.

10 CFR 50.55a(a)(1)(i), "ASME Boiler and Pressure Vessel Code, Section III," lists the documents approved for incorporation by reference including the NRC-approved editions and addenda for ASME B&PV Code Section III (excluding Nonmandatory Appendices).

10 CFR 50.55a(b), "Use and conditions on the use of standards," requires that systems and components of boiling water-cooled nuclear power reactors must meet the requirements of the ASME B&PV Code and the ASME OM Code as specified in this paragraph (b).

10 CFR 50.55a(z), "Alternatives to codes and standards requirements," allows the use of proposed alternatives to the ASME B&PV Code (i.e., the requirements of paragraphs (b) through (h) of 10 CFR 50.55a or portions thereof) when authorized by the Director, Office of Nuclear Reactor Regulation. A proposed alternative must be submitted and authorized prior to implementation. The applicant or licensee must demonstrate that:

1. Acceptable level of quality and safety. The proposed alternative would provide an acceptable level of quality and safety; or
2. Hardship without a compensating increase in quality and safety. Compliance with the specified requirements of this section would result in hardship or unusual difficulty without a compensating increase in the level of quality and safety.

10 CFR 50.65, "Requirements for monitoring the effectiveness of maintenance at nuclear power plants," requires monitoring of the performance or condition of SSCs, against licensee-established goals, in a manner sufficient to provide reasonable assurance that these structures, systems, and components, as defined in paragraph (b) of this section, are capable of fulfilling their intended functions. These goals shall be established commensurate with safety and, where practical, take into account industrywide operating experience. When the performance or condition of an SSC does not meet established goals, appropriate corrective action shall be taken. This includes structures monitoring and maintenance requirements for seismic Category I structures including the integrated RB.

10 CFR 50 Appendix A, General Design Criterion (GDC) 1, "Quality standards and records," requires that SSCs important to safety shall be designed, fabricated, erected, and tested to quality standards commensurate with the importance of the safety functions to be performed. Where generally recognized codes and standards are used, they shall be identified and evaluated to determine their applicability, adequacy, and sufficiency and shall be supplemented or modified as necessary to assure a quality product in keeping with the required safety function. A quality assurance program shall be established and implemented in order to provide adequate assurance that these SSCs will satisfactorily perform their safety functions. Appropriate records of the design, fabrication, erection, and testing of SSCs important to safety shall be maintained by or under the control of the nuclear power unit licensee throughout the life of the unit.

10 CFR 50 Appendix A, GDC 2, "Design bases for protection against natural phenomena," requires that SSCs important to safety shall be designed to withstand the effects of natural phenomena such as earthquakes, tornadoes, hurricanes, floods, tsunami, and seiches without loss of capability to perform their safety functions. The design bases for these SSCs shall reflect: (1) Appropriate consideration of the most severe of the natural phenomena that have been historically reported for the site and surrounding area, with sufficient margin for the limited accuracy, quantity, and period of time in which the historical data have been accumulated, (2) appropriate

combinations of the effects of normal and accident conditions with the effects of the natural phenomena and (3) the importance of the safety functions to be performed.

10 CFR 50 Appendix A, GDC 4, "Environmental and dynamic effects design bases," requires that SSCs important to safety shall be designed to accommodate the effects of and to be compatible with the environmental conditions associated with normal operation, maintenance, testing, and postulated accidents, including loss-of-coolant accidents. These SSCs shall be appropriately protected against dynamic effects, including the effects of missiles, pipe whipping, and discharging fluids, that may result from equipment failures and from events and conditions outside the nuclear power unit. However, dynamic effects associated with postulated pipe ruptures in nuclear power units may be excluded from the design basis when analyses reviewed and approved by the Commission demonstrate that the probability of fluid system piping rupture is extremely low under conditions consistent with the design basis for the piping.

10 CFR 50 Appendix A, GDC 16, "Containment design," requires that reactor containment and associated systems shall be provided to establish an essentially leak-tight barrier against the uncontrolled release of radioactivity to the environment and to assure that the containment design conditions important to safety are not exceeded for as long as postulated accident conditions require.

10 CFR 50 Appendix A, GDC 50, "Containment design basis," requires that the reactor containment structure, including access openings, penetrations, and the containment heat removal system shall be designed so that the containment structure and its internal compartments can accommodate, without exceeding the design leakage rate and with sufficient margin, the calculated pressure and temperature conditions resulting from any loss-of-coolant accident. This margin shall reflect consideration of (1) the effects of potential energy sources which have not been included in the determination of the peak conditions, such as energy in steam generators and as required by 10 CFR 50.44 energy from metal-water and other chemical reactions that may result from degradation but not total failure of emergency core cooling functioning, (2) the limited experience and experimental data available for defining accident phenomena and containment responses, and (3) the conservatism of the calculational model and input parameters.

10 CFR 50 Appendix A, GDC 51, "Fracture prevention of containment pressure boundary," requires that the reactor containment boundary shall be designed with sufficient margin to assure that under operating, maintenance, testing, and postulated accident conditions (1) its ferritic materials behave in a nonbrittle manner and (2) the probability of rapidly propagating fracture is minimized. The design shall reflect 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.

10 CFR 50 Appendix A, GDC 53, "Provisions for containment testing and inspection," requires that the reactor containment shall be 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 leaktightness of penetrations which have resilient seals and expansion bellows.

10 CFR 50 Appendix J, "Primary reactor containment leakage testing for water-cooled power reactors," requires that primary reactor containments shall meet the containment leakage test

requirements set forth in this appendix. These test requirements provide for preoperational and periodic verification by tests of the leak-tight integrity of the primary reactor containment, and systems and components which penetrate containment of water-cooled power reactors and establish the acceptance criteria for these tests. The purposes of the tests are to assure that (a) leakage through the primary reactor containment and systems and components penetrating primary containment shall not exceed allowable leakage rate values as specified in the technical specifications or associated bases; and (b) periodic surveillance of reactor containment penetrations and isolation valves is performed so that proper maintenance and repairs are made during the service life of the containment, and systems and components penetrating primary containment. These test requirements may also be used for guidance in establishing appropriate containment leakage test requirements in technical specifications or associated bases for other types of nuclear power reactors.

10 CFR Part 50, Appendix S, "Earthquake engineering criteria for nuclear power plants," requires that for Safe Shutdown Earthquake (SSE) ground motions, SSCs will remain functional and within applicable stress, strain, and deformation limits. The required safety functions of SSCs must be assured during and after the vibratory ground motion through design, testing, or qualification methods. The evaluation must take into account Soil-Structure Interaction (SSI) effects and the expected duration of the vibratory motion. Also requires that the horizontal component of the SSE ground motion in the free field at the foundation level of the structures must be an appropriate response spectrum with a Peak Ground Acceleration (PGA) of at least 0.1g.

4.1.3 NRC Regulatory Guidance

NUREG-0800, Standard Review Plan (SRP) 3.7.1, "Seismic Design Parameters," Revision 4, provides review guidance to the NRC Staff responsible for seismic and structural analysis reviews and secondarily for review of seismic ground motion development. This includes relevant specific areas of review including the following:

- earthquake design ground motions that are exerted on the plant structures and used in SSI analyses, including Operating Basis Earthquake (OBE) and SSE
- percentage of critical damping values used for the seismic analysis of seismic Category I SSCs for both the OBE and the SSE
- description of the supporting media for each seismic Category I structure, including foundation embedment depth, depth of soil over bedrock, soil layering characteristics, highest groundwater elevation, dimensions of the structural foundation, total structural height, topographical conditions of the sites, and soil properties (including strain-dependent properties) and their assumed variability to permit evaluation of the applicability of continuum, finite-element or lumped-spring approaches for SSI analysis

NUREG-0800, SRP 3.7.2, "Seismic System Analysis," Revision 4, provides review guidance to the NRC Staff responsible for seismic and structural analysis reviews. This includes relevant specific areas of review including the following:

- seismic analysis methods
- significant natural frequencies and responses
- criteria and procedures used in models for the seismic system analyses

- SSI analyses
- procedures and methods for developing in-structure response spectra
- procedures by which the three components of design ground motion (time history or response spectra) are considered
- modes with natural frequencies greater than the frequency at which the spectral acceleration converges to approximately the Zero Period Acceleration (ZPA) frequency if applicable
- design criteria to account for the seismic motion of non-seismic Category I structures (or portions thereof) in the seismic design of seismic Category I structures (or portions thereof)
- procedures that are used to consider the effects of the expected variations of structural properties, critical damping values, soil properties, and SSI on the floor response spectra and response time histories
- justification for the use of equivalent vertical static load factors in calculating the vertical response loads for designing seismic Category I SSCs in lieu of the use of a vertical seismic system dynamic analysis is reviewed if applicable
- method employed to consider torsional effects in the seismic analysis of seismic Category I structures
- comparison of seismic responses for major seismic Category I structures using modal response spectrum and time history approaches is reviewed if applicable
- procedure employed to account for different critical damping values in different elements of the system structural model
- description of the method and procedure used to determine design seismic overturning moments and sliding forces, structure to soil pressures beneath the foundation and alongside walls (including potential sliding and separation effects) and soil friction for all seismic Category I structures

NUREG-0800, SRP 3.7.3, "Seismic Subsystem Analysis Review Responsibilities," Revision 4, provides review guidance to the NRC Staff responsible for seismic and structural analysis reviews. This includes relevant specific areas of review similar to that described in Subsection I.1 of SRP 3.7.2 but as applied to seismic Category I subsystems.

NUREG-0800, SRP 3.8.1, "Concrete Containment," Revision 4, provides review guidance to the NRC Staff responsible for structural analysis reviews for concrete containments. Although this SRP section is not directly applicable to the design of the SCCV, the guidance is reviewed to determine what remains relevant for the NRC Staff to consider in their review. This includes specific areas of review including the following:

- descriptive information, including plans and sections of the containment structure, to establish that sufficient information is provided to define the primary structural aspects and elements relied upon to perform the containment function, including structural and functional characteristics

- design codes, standards, specifications, regulations, and RGs and other industry standards that are applied in the design fabrication, construction, testing, and inservice surveillance of the containment
- information pertaining to the applicable design loads and various combinations thereof, with emphasis on the extent of compliance with ASME B&PV Code requirements
- design and analysis procedures used for the containment with emphasis on the extent of compliance with ASME B&PV Code requirements
- design limits imposed on the various parameters that quantify the structural behavior of the containment, with emphasis on the extent of compliance with ASME B&PV Code requirements
- materials that are used in construction of the containment with emphasis on the extent of compliance with ASME B&PV Code requirements
- quality control program that is proposed for the fabrication and construction of the containment with emphasis on the extent of compliance with ASME B&PV Code requirements
- any special design provisions (e.g., providing sufficient physical access, providing alternative means for identification of conditions in inaccessible areas that can lead to degradation, or providing remote visual monitoring of high-radiation areas) to accommodate inservice inspection
- preoperational structural testing program for the completed containment and for individual components, such as personnel and equipment locks and hatches, and includes the objectives of the test program and acceptance criteria, with emphasis on the extent of compliance with ASME B&PV Code requirements, including inservice surveillance programs

NUREG-0800, SRP 3.8.2, "Steel Containment," Revision 3, provides review guidance to the NRC Staff responsible for structural analysis reviews for steel containments. Although this SRP section is not directly applicable to the design of the SCCV, the guidance is reviewed to determine what remains relevant for the NRC Staff to consider in their review. This includes specific areas of review including the following:

- descriptive information, including plans and sections of the containment structure, to establish that sufficient information is provided to define the primary structural aspects and elements relied upon to perform the containment function, including structural and functional characteristics, including steel components of concrete containments that resist pressure and are not backed by structural concrete (e.g., the containment head in a BWR)
- design codes, standards, specifications, regulations, and RGs and other industry standards that are applied in the design fabrication, construction, testing, and inservice surveillance of the containment
- information pertaining to the applicable design loads and various combinations thereof, with emphasis on the extent of compliance with ASME B&PV Code requirements
- design and analysis procedures used for the containment with emphasis on the extent of compliance with ASME B&PV Code requirements

- design limits imposed on the various parameters that quantify the structural behavior of the containment, with emphasis on the extent of compliance with ASME B&PV Code requirements
- materials that are used in construction of the containment with emphasis on the extent of compliance with ASME B&PV Code requirements
- quality control program that is proposed for the fabrication and construction of the containment with emphasis on the extent of compliance with ASME B&PV Code requirements, including nondestructive examination of the materials, including tests to determine their physical properties, welding procedures, and erection tolerances
- any special construction techniques, if proposed, to determine their effects on the structural integrity of the completed containment
- preoperational structural testing program for the completed containment and for individual components, such as personnel and equipment locks and hatches, and includes the objectives of the test program and acceptance criteria, with emphasis on the extent of compliance with ASME B&PV Code requirements, including inservice surveillance programs
- special testing and inservice surveillance requirements proposed for new or previously untried design approaches, and for new reactors, it is important to accommodate inservice inspection of critical areas

NUREG-0800, SRP 3.8.3, "Concrete and Steel Internal Structures of Steel or Concrete Containments," provides review guidance to the NRC Staff responsible for structural analysis reviews. This includes specific areas of review including the following:

- descriptive information, including plans and sections of the various internal structures, to establish that sufficient information is provided to define the primary structural aspects and elements relied upon to perform the safety-related functions of these structures
- capability of the internal structures to resisting loads and load combinations to which they may be subjected and should not become the initiator of a loss-of-coolant accident (LOCA), with the structures able to mitigate its consequences by protecting the containment and other engineered safety features from the accident's effects such as jet forces and whipping pipes
- plant designs may also use modular construction methods for the major containment internal structures, with wall modules typically constructed from large, prefabricated sections of steel plates spaced apart with intermittent steel members, joined with other modules at the site, and then filled with concrete, and with the concrete fill used in wall modules either structural concrete with reinforcement (composite construction) or fill concrete of low strength without reinforcement, or heavy concrete for radiation shielding, and with floor modules consisting of prefabricated steel members and plates combined with poured concrete to create a composite section, and the structural module design, fabrication, configuration, layout, and connections may be reviewed on a case-by-case basis

- design codes, standards, specifications, and RGs, as well as industry standards, that are applied in the design, fabrication, construction, testing, and surveillance of the containment structures
- applicable design loads and associated load combinations
- design and analysis procedures used for the containment internal structures, with an emphasis on the extent of compliance with the applicable codes as indicated in Subsection II.2 of this SRP
- design limits imposed on the various parameters that quantify the structural behavior of the various interior structures of the containment, particularly with respect to stresses, strains, deformations, and factors of safety against structural failure, with emphasis on the extent of compliance with the applicable codes indicated in Subsection II.5 of this SRP
- materials that are used in the construction of the containment internal structures, including concrete ingredients, reinforcing bars and splices, structural steel, and various supports and anchors are among the major materials of construction
- quality control program proposed for the fabrication and construction of the containment internal structures, including nondestructive examination of the materials to determine physical properties, placement of concrete, and erection tolerances
- special, new, or unique construction techniques, such as the use of modular construction methods, if used, may be reviewed on a case-by-case basis to determine their effects on the structural integrity of the completed containment internal structures
- for seismic Category I structures inside containment, information on structures monitoring and maintenance requirements including inservice inspection of critical areas, special design provisions (e.g., sufficient physical access, alternative means for identification of conditions in inaccessible areas that can lead to degradation, remote visual monitoring of high radiation areas) to accommodate inservice inspection of containment internal structures, and post-construction testing and inservice surveillance programs for containment internal structures such as periodic examination of inaccessible areas

NUREG-0800, SRP 3.8.4, “Other Seismic Category I Structures,” provides review guidance to the NRC Staff responsible for structural analysis reviews. This includes specific areas of review that are applicable to the RB structure including the following:

- descriptive information, including plans and sections of each structure, to establish that there is sufficient information to define the primary structural aspects and elements relied upon for the structure to perform the intended safety function, and the relationship between adjacent structures, including the separation provided or structural ties, if any
- design codes, standards, specifications, RGs, and other industry standards that are applied in the design, fabrication, construction, testing, and surveillance of seismic Category I structures
- applicable design loads and various load combinations
- design and analysis procedures used for seismic Category I structures focusing on the extent of compliance with American Concrete Institute (ACI) 349, with supplemental

guidance by RG 1.142 for concrete structures and ANSI/AISC N690-1994 including Supplement 2 (2004) for steel structures

- design limits imposed on the various parameters that serve to quantify the structural behavior of each structure and its components, with specific attention to stresses, strains, gross deformations, and factors of safety against structural failure, and for each load combination specified, the allowable limits compared with the acceptable limits delineated in Subsection II.5 of this SRP section
- materials used in the construction of seismic Category I structures, including concrete ingredients, the reinforcing bars and splices, and the structural steel and anchors
- quality control parameters that are proposed for the fabrication and construction of seismic Category I structures, including nondestructive examination of the materials to determine physical properties, placement of concrete, and erection tolerances
- special construction techniques, such as modular construction methods, if used, may be reviewed on a case-by-case basis to determine their effects on the structural integrity of the completed structure
- for seismic Category I structures outside containment, information on structures monitoring and maintenance requirements including accommodation for inservice inspection of critical areas, any special design provisions (e.g., providing sufficient physical access, providing alternative means for identification of conditions in inaccessible areas that can lead to degradation, remote visual monitoring of high radiation areas) to accommodate inservice inspection of other seismic Category I structures, and post-construction testing and inservice surveillance programs for other seismic Category I structures, such as periodic examination of inaccessible areas, monitoring of ground water chemistry, and monitoring of settlements and differential displacements, may be reviewed on a case-by-case basis
- masonry walls, if applicable, should include, at a minimum, those items identified in Appendix A to this SRP section

Additional guidance may be considered if relevant to the specific use of SC materials for the integrated RB including the following:

- RG 1.7, “Control of Combustible Gas Concentrations in Containment Following a Loss-of-Coolant Accident”
- RG 1.57, “Design Limits and Loading Combinations for Metal Primary Reactor Containment System Components”
- RG 1.60, “Design Response Spectra for Seismic Design of Nuclear Power Plants”
- RG 1.61, “Damping Values for Seismic Design of Nuclear Power Plants”
- RG 1.69, “Concrete Radiation Shields for Nuclear Power Plants”
- RG 1.70, “Standard Format and Content of Safety Analysis Reports for Nuclear Power Plants”
- RG 1.84, “Design, Fabrication, and Materials Code Case Acceptability, ASME Section III”

- RG 1.91, "Evaluations of Explosions Postulated to Occur on Transportation Routes Near Nuclear Power Plants"
- RG 1.92, "Combining Modal Responses and Spatial Components in Seismic Response Analysis"
- RG 1.115, "Protection Against Low Trajectory Turbine Missiles"
- RG 1.122, "Development of Floor Design Response Spectra for Seismic Design of Floor-Supported Equipment or Components"
- RG 1.127, "Inspection of Water-Control Structures Associated with Nuclear Power Plants"
- RG 1.136, "Materials, Construction, and Testing of Concrete Containments"
- RG 1.142, "Safety-Related Concrete Structures for Nuclear Power Plants (other than Reactor Vessels and Containments)"
- RG 1.147, "Inservice Inspection Code Case Acceptability, ASME Section XI, Division 1"
- RG 1.160, "Monitoring the Effectiveness of Maintenance at Nuclear Power Plants"
- RG 1.193, "ASME Code Cases Not Approved for Use"
- RG 1.199, "Anchoring Components and Structural Supports in Concrete"
- RG 1.221, "Design-Basis Hurricane and Hurricane Missiles for Nuclear Power Plants"
- RG 1.243, "Safety Related Steel Structures and Steel-Plate Composite Walls for Other Than Reactor Vessels and Containments"
- RG 1.132, "Site Investigations for Foundations of Nuclear Power Plants"
- RG 1.138, "Laboratory Investigations of Soils and Rocks for Engineering Analysis and Design of Nuclear Power Plants"

4.2 Licensing Topical Report (LTR) NEDC-33926P Content

LTR NEDC-33926P (Reference 5-10) includes specific BWRX-300 design parameters to address design and construction requirements for the integrated RB, including but not limited to the following technical evaluation information:

- Description of the integrated RB consisting of the Containment structure, including SCCV and containment internal structures, and the RB structure
- Design parameters
- Description of the seismic and structural analysis approach
- Loads and load combinations
- Allowable materials
- Description of finite element model of the integrated RB
- Effective stiffness of SC members
- Description of model validation and verification
- Design requirements

- Acceptance criteria for SCCV
- Impulsive and impactive design of SC including Steel Bricks™
- Design and detailing requirements around large Containment penetration and openings
- Design and detailing requirements for critical SC module connections
- Fabrication and construction requirements, including welded construction using Steel Bricks™
- Construction testing and examination requirements, including weld examination and qualification for Steel Bricks™
- Pre-service and In-service inspection and testing requirements
- Integrated Leak Rate Test (ILRT) and Structural Integrity Test (SIT) requirements
- Ultimate pressure capacity determination
- Additional design considerations, including:
  - Acceptability of ANSI/AISC N690 for all structural elements, including slabs, for the design of the integrated RB
  - Any exceptions to ANSI/AISC N690 requirements for Steel Bricks™
  - Effect of curvature on behavior of BWRX-300 SC (including Steel Bricks™) structures

In addition to the above technical evaluation, the LTR is planned to contain a regulatory evaluation of the basis for acceptability considering both CNSC and NRC regulatory requirements and guidance.

5.0 REFERENCES

5-1 Modular Walling Systems, https://www.steelbricks.co.uk/SteelBricks.aspx.

5-2 AISC Design Guide 32, Design of Modular Steel-Plate Composite Walls for Safety-Related Nuclear Facilities, 2017.

5-3 ASME B&PV Code, Section III, Rules for Construction of Nuclear Facility Components.

5-4 CSA N291, Requirements for Safety-Related Structures for Nuclear Power Plants, CSA Group, 2019.

5-5 CSA N287 series of documents, including CSA N287.3, Design Requirements for Concrete Containment Structures for Nuclear Power Plants, CSA Group, 2014 (R2019).

5-6 CSA N289 series of documents, CSA Group.

5-7 ANSI/AISC N690-18, Specification for Safety-Related Steel Structures for Nuclear Facilities, American Institute of Steel Construction.

5-8 Regulatory Guide 1.243, Safety Related Steel Structures and Steel-Plate Composite Walls for Other Than Reactor Vessels and Containments, U.S. Nuclear Regulatory Commission, 2021.

5-9 ASCE/SEI 43, Seismic Design Criteria or Structures, Systems, and Components in Nuclear Facilities, American Society of Civil Engineers, 2019.

5-10 Licensing Topical Report NEDC-33926P, BWRX-300 Steel-Plate Composite (SC) Containment Vessel (SCCV) and Reactor Building Structural Design.

5-11 CNSC REGDOC-1.1.2, Version 2 (Draft) "Reactor Facilities, Licence application Guide: Licence to Construct a Reactor Facility", November 2021.

5-12 CNSC REGDOC-2.5.2, Version 1 "Physical Design – Design of Reactor Facilities," May 2014.

5-13 CNSC REGDOC-2.6.3, "Aging Management," March 2014.