Joint Report on GEH BWRX-300 Steel-Plate Composite Containment Vessel and Reactor Building Structural Design Topical Report

A Collaborative Review by the U.S. Nuclear Regulatory Commission and the Canadian Nuclear Safety Commission

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List of Acronyms

ACI	American Concrete Institute
ADAMS	Agencywide Documents Access and Management System
AISC	American Institute of Steel Construction
ANSI	American National Standards Institute
ART-SMR	Advanced Reactor Technologies and Small Modular Reactors
ASCE	American Society of Civil Engineers
ASME	American Society of Mechanical Engineers
B&PV	boiler and pressure vessel
BWR	boiling-w ater reactor
CFR	*Code of Federal Regulations*
CNSC	Canadian Nuclear Safety Commission
CPA	construction permit application
CSA	Canadian Standards Association
DBE	design-basis earthquake
DP	diaphragm plate
DP-SC	diaphragm plate-steel composite
GDC	general design criteria
GEH	GE-Hitachi Nuclear Energy Americas LLC
IP	in-plane
LTC	License To Construct
LTR	licensing topical report
MOC	memorandum of cooperation
MOU	memorandum of understanding
NPP	nuclear power plant
NRC	Nuclear Regulatory Commission
NRIC	National Reactor Innovation Center
OOP	out-of-plane
OPG	Ontario Power Generation
RAI	request for additional information
RB	reactor building
REGDOC	Regulatory Document
RG	Regulatory Guide
RPV	reactor pressure vessel
SC	steel-plate composite
SCCV	steel-plate composite containment vessel
SEI	Structural Engineering Institute
SER	safety evaluation report
SMR	small modular reactor
SRP	Standard Review Plan
SSC	structure, system, and component
TVA	Tennessee Valley Authority
US	United States
VDR	vendor design review

Executive Summary

In August 2019, the Canadian Nuclear Safety Commission (CNSC) and the U.S. Nuclear Regulatory Commission (NRC) signed a Memorandum of Cooperation (MOC) to increase collaboration on technical reviews of small modular reactor (SMR) and advanced reactor technologies. In 2023, NRC and CNSC staff performed a review of a GE-Hitachi Nuclear Energy Americas, LLC (GEH) white paper on the BWRX-300 Steel-Plate Composite Containment Vessel and Reactor Building Structural Design, and the review results were documented in a joint report [23]. Subsequent to the white paper, GEH submitted licensing topical report (LTR) NEDO-33926 to both regulators. This joint review report is based on revision 2 of the LTR. NRC and CNSC staff engaged in a collaborative technical review of the LTR. The results of the collaborative technical review have been used by each regulator to inform their regulatory processes and activities. In the U.S., the LTR review is a pre-licensing activity performed as per NRC internal office instruction LIC-500, "Topical Review Process". In Canada, the LTR was included in the existing License to Construct (LTC) Application by the applicant Ontario Power Generation Inc. (OPG) and communication on the LTR shifted from being with GEH (and OPG as observer) during the audit process to being with OPG so that it is included in the licence application process.

The LTR review results from the joint CNSC and NRC technical collaboration establishes that the LTR provides an acceptable way of meeting regulatory requirements and expectations for both regulators. The LTR, as a methodology document, provides sufficient information on the approach to meet design requirements for diaphragm plate-steel composite (DP-SC) modules. In addition, the LTR is consistent with design codes (where available) and provides further justification where code design requirements are not present through other relevant materials (draft design code, test results, research results). CNSC and NRC staff identified several areas where further elaboration of the design approach is needed as part of design development (e.g. in detailed design) in the form of limitations and conditions for the NRC, and comments for CNSC.

Introduction

This report documents the joint review activities between the CNSC and the NRC regarding GE-Hitachi Nuclear Energy Americas LLC's (GEH) Steel-Plate Composite (SC) Containment Vessel (SCCV) and Reactor Building (RB) Structural Design LTR for the BWRX-300. This report is based on revision 2 of the LTR (NRC Agencywide Documents Access and Management System (ADAMS) Accession No. ML24110A132, CNSC e-doc # 7268622) [22]. The results of this report may be used by the vendor or an applicant in future discussions with either regulator, but they are not legally binding on the CNSC or the NRC.

In April 2022, the Tennessee Valley Authority (TVA) and Ontario Power Generation Inc. (OPG) announced plans to jointly work to help develop and deploy SMRs in both Canada and the U.S. They have signed a memorandum of understanding (MOU) that allows the companies to coordinate efforts on the design, licensing, construction, and operation of SMRs. The CNSC and NRC are currently engaged in licensing and pre-application activities with OPG and TVA, respectively, in preparation to build the BWRX-300 reactor in Canada and the U.S. In September of 2022, CNSC and NRC signed a charter [15] establishing a collaborative relationship on the BWRX-300 SMR design project. Under this charter, OPG, TVA and GEH will identify licensing topics for consideration by the CNSC and NRC for cooperative reviews. Further, OPG, TVA, and GEH will identify challenges with applying existing guidance or frameworks, provide technical information to facilitate timely and efficient safety reviews, and ensure efficient communication with both regulators.

Nothing in this report fetters the powers, duties or discretion of CNSC or NRC designated officers, CNSC or NRC inspectors or the respective Commissions regarding making regulatory decisions or taking regulatory action. Nothing in this report is to be construed or interpreted as affecting the jurisdiction and discretion of the CNSC in any assessment of any application for licensing purposes under the Nuclear Safety and Control Act (NSCA) of 2000, last amended in 2017 [5], its associated regulations or the CNSC Rules of Procedure. Likewise, nothing in this report is to be construed or interpreted as affecting the jurisdiction and discretion of the NRC in any evaluation of any application for licensing purposes under the Atomic Energy Act of 1954, as amended, its associated regulations and the NRC Management Directives. This report does not involve the issuance of a license under Section 24 of the NSCA (Canada) or under Section 103 of the Atomic Energy Act of 1954 (USA). The conclusions in this collaborative report are of the CNSC and NRC staff.

Background on the Memorandum of Understanding

In August 2017, the CNSC and NRC signed a MOU [2]. Subsequently, in August 2019 the CNSC and NRC signed a joint MOC [3] aimed at enhancing technical reviews of advanced reactor and SMR technologies.

CNSC/NRC cooperation provides opportunities for both agencies to share scientific information about technical matters that could support more efficient and streamlined reviews of advanced reactor and SMR technologies in the future. Cooperative activities are conducted with acknowledgement of differences in the regulatory frameworks and licensing processes of Canada and the United States, while leveraging fundamental scientific and engineering findings common to each country to the extent practicable. The cooperative activities under the MOU and MOC are intended to:

- contribute to better use of regulator's resources by leveraging the technical knowledge and resources between the NRC and the CNSC

- enhance the depth and breadth of understanding of the respective staff of the CNSC and NRC on the counterpart nation's regulatory review activities and requirements

- enhance the joint opportunities for learning and understanding the advanced reactor and SMR technologies being reviewed

Activities under the MOC are coordinated by the CNSC/NRC Advanced Reactor Technologies and Small Modular Reactors (ART-SMR) Committee. This committee approves and prioritizes work plans to accomplish specific cooperative activities under the MOC. The committee functions according to the terms of reference established in 2020 [4].

This report documents the cooperative review activities performed on GEH's methodology for constructing the BWRX-300 reactor design's containment vessel and RB using diaphragm plate-steel composite (DP-SC) modules.

GEH Engagement with the Regulators

GEH Engagement with the NRC

In September 2019, GEH initiated pre-application activities with the NRC on its BWRX-300 reactor design to support a future license application. The BWRX-300 is a ~300 MWe light-water-cooled, natural circulation boiling-water SMR with passive safety systems. The design of the BWRX-300 is based in part on the U.S. NRC-certified 1,520 MWe Economic Simplified Boiling-Water Reactor. GEH has submitted six LTRs on key licensing issues since 2020 that have been reviewed and approved by the NRC staff. LTRs describe the BWRX-300 design approaches and analyses methodologies for the BWRX-300 SMR in advance of a future licensing application under Title 10 of the *Code of Federal Regulations* (10 CFR) Part 50, "Domestic Licensing of Production and Utilization Facilities," or Part 52, "Licenses, Certifications, and Approvals for Nuclear Power Plants." GEH also submitted two white papers in 2022 as part of pre-LTR submittal engagements, including one on containment and RB structural design, which is related to the subject of this report. On May 4, 2023, GEH submitted the subject LTR for NRC review and over the course of the review, revised the LTR twice before this report was written. These revisions were made in response to a joint regulatory audit [19] and NRC requests for additional information [20]. On August 19, 2024, the NRC issued its safety evaluation report (SER) [21]. The SER states the detailed bases for the NRC conclusions regarding the LTR set forth in this report. (It is noted that subsequent to submittal of the LTR, GEH, became GE Vernova, transitioning its nuclear arm to GE Vernova Hitachi Nuclear Energy (GVH) as part of this new entity. This joint report refers to GEH because the incoming LTR was submitted under the name of GEH.)

GEH Engagement with the CNSC

GEH and the CNSC entered a combined optional pre-licensing Phase 1 and 2 Vendor Design Review (VDR) regarding the BWRX-300 reactor in December 2019 through a service agreement. The VDR examined whether GEH understands CNSC regulatory requirements and to what extent the reactor design meets those requirements. To provide GEH with early feedback, CNSC carried out the VDR during the ongoing design process while the GEH's reactor design was still evolving. This allowed for the early identification and resolution of potential regulatory or technical issues in the safety analysis and design process, particularly those that could result in significant changes to GEH's design.

The BWRX-300 VDR concluded that, based on the review of the design features reviewed during the VDR, GEH understands and correctly interpreted the intent of Canadian regulatory requirements for the design of nuclear power plant (NPPs). However, the review did reveal some technical areas for development to better demonstrate adherence to CNSC requirements.

GEH has been identified as the design authority for the BWRX-300 OPG's LTC application, submitted to the CNSC in October 2022. The LTC application includes the use of DP-SC for the RB and containment structures and for the concrete pedestal, supporting the RPV. In March 2025, the CNSC Commission issued a power reactor construction license to OPG, following a 2-part public hearing, including the use of DP-SC modules as identified in the application.

Scope and Objectives for the Cooperative Activity

The work plan [3] issued in January 2024, describes the scope of work and objectives as follows:

> *"To share regulatory experiences and insights for the BWRX-300 SMR design. Specifically, the scope of work is to perform a collaborative review of Licensing Topical*

Report NEDC-33926P, BWRX-300 Steel-Plate Composite Containment Vessel and Reactor Building Structural Design, which includes design and construction methodologies, regulatory approaches, and treatment of unique aspects of the BWRX-300 structural design."

The main objective of this report is to document the USNRC and the CNSC staff's joint assessment of the BWRX-300 SCCV and RB Structural Design LTR.

Regulatory Framework

CNSC

The CNSC's regulatory framework provides regulatory instruments that clearly state CNSC's regulatory requirements and expectations, and guidance material. All applicants and licensees are required to comply with the NSCA and regulations, as well as any other statutory and regulatory instruments. The CNSC has also developed and published regulatory documents (REGDOC) that clearly provide expectations for compliance with the NSCA, regulations and other standards the CNSC has included in its framework (i.e., CSA Group and ASME standards). REGDOC provisions are those that an applicant or licensee must meet. However, an applicant or licensee may propose alternative means to meet the REGDOCs, by demonstrating with supported evidence that the intent of a requirement is satisfied in a safety-neutral or more safe manner. Allowing for alternative approaches to meet requirements demonstrates that the CNSC's regulatory framework provides flexibility for licensees and applicants when proposing novel technology or design features. Alternative approaches are further discussed in the following sections.

The CNSC generally adopts a risk-informed, performance based regulatory approach; therefore, it does not prescribe specific codes and standards for the design and manufacturing of systems, structures, and components (SSCs). However, it requires that the quality of SSCs be commensurate with their safety classification for design, construction, and maintenance.

The most relevant REGDOCs to GEH's use of DP-SC in the containment and RB structures are presented below:

<u>CNSC Regulatory Documents</u>

CNSC REGDOC-1.1.2, "License Application Guide: Guide to Construct A Reactor Facility" [17]

This document outlines the requirements and expectations for a license application to construct a reactor facility. It contains specific information requirements and expectations for civil structures (that includes RB) and for the pressure boundary (containment part of DP-SC structure).

CNSC REGDOC-2.5.2, "Design of Reactor Facilities: Nuclear Power Plants" [6]

This document outlines the requirements and expectations for the design of reactor facilities, including, but not limited to requirements for: identification of codes and standards, identification of hazards (natural and man-made), design-basis and design extension conditions, initiating events, seismic and environmental qualification, containment requirements, overpressure protection.

CNSC REGDOC-2.6.3, “Fitness for Service Aging Management” [18]

This document outlines the requirements for aging management of structures, systems and components of NPPs. It requires that aging management be considered during the design phase.

Other Canadian Design Standards

The CNSC regulatory framework also includes other Canadian design standards, including CSA Group standards.

The CSA N287 series [7] of standards provides requirements for concrete containment structures and some of its requirements would be applicable to the SCCV.

CSA N291, “Requirements for nuclear safety-related structures,” provides the design requirements for safety-related structures, other than containment [8], e.g. RB (when RB is a separate structure).

Existing Canadian design standards do not explicitly consider SC material of the type and extent proposed to be used in the LTR. Recent Canadian steel design code CSA S16-24, published in 2024, included an informative appendix on SC walls and CNSC staff are considering it in their review. Although CSA N287 series and N291 are listed as guidance in REGDOC-2.5.2, the CNSC reviewed DP-SC structures in the LTR and OPG’s application in accordance with their compliance with the CSA N287 series and N291.

The basis of the review to allow it to move forward within the bounds of the Canadian Regulatory framework, in the absence of specific design codes, stems from the use of alternative approach as defined in CNSC REGDOC 2.5.2, namely:

“*9. Alternative approaches*

The requirements in this regulatory document are intended to be technology neutral for water-cooled reactor designs. It is recognized that specific technologies may use alternative approaches.

The CNSC will consider alternative approaches to the requirements in this document 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
3. the application of the requirements in this document would not serve the underlying purpose, or is not necessary to achieve the underlying purpose
Any alternative approach shall demonstrate equivalence to the outcomes associated with the use of the requirements set out in this regulatory document.”

In the evolution of their LTC application, OPG has submitted additional information aiming to demonstrate how they meet Canadian codes and standards, including some of the above, to allow for use of available Canadian nuclear quality assurance and construction approaches. This information is not part of the joint review with NRC and is only mentioned here for completeness.

USNRC

New reactor designs in the U.S. are assessed against the criteria outlined in 10 CFR Part 50 [9] or 10 CFR Part 52 [10] although the substantive regulations are largely equivalent. Additionally, the NRC is currently developing another licensing pathway for new reactor designs: 10 CFR Part 53, “Risk-Informed, Technology-Inclusive Regulatory Framework for Commercial Nuclear Plants.”

GEH had entered into an agreement with TVA to develop a construction permit application under 10 CFR Part 50 to potentially deploy a BWRX-300 at the Clinch River site. As such, the NRC has focused on the regulatory requirements in Part 50 to assess the design.

For the BWRX 300 integrated RB, the SCCV, and other internal structures, the applicable NRC regulations include:

- 10 CFR 50.34, “Contents of applications; technical information,”
- 10 CFR 50.44, “Combustible gas control for nuclear power reactors,”
- 10 CFR 50.55a, “Codes and Standards,”
- 10 CFR 50.65, “Requirements for monitoring the effectiveness of maintenance at nuclear power plants,”
- 10 CFR 50.150 “Aircraft Impact Assessment,”
- 10 CFR Part 50, Appendix A, “General Design Criteria for Nuclear Power Plants (GDCs)”, including: GDC 1, “Quality standards and records,” GDC 2, “Design bases for protection against natural phenomena,” GDC 4, “Environmental and dynamic effects design bases,” GDC 16, “Containment design,” GDC 50, “Containment design-basis, GDC 51, “Fracture prevention of reactor coolant pressure boundary,” GDC 52, “Capability for containment leakage rate testing,” and GDC 53, “Provisions for containment testing and inspection,”
- 10 CFR Part 50, Appendix B, “Quality Assurance Criteria for Nuclear Power Plants and Fuel Reprocessing Plants,”
- 10 CFR Part 50, Appendix J, “Primary Reactor Containment Leakage Testing for Water-Cooled Power Reactors,” and
- 10 CFR Part 50, Appendix S, “Earthquake Engineering Criteria for Nuclear Power Plants.”

Applicable NRC guidance, including regulatory guides (RGs) are as follows:

- NUREG-0800, “Standard Review Plan for the Review of Safety Analysis Reports for Nuclear Power Plants” (SRP),
- RG 1.7, “Control of Combustible Gas Concentrations in Containment,”

- RG 1.26, “Quality Group Classifications and Standards for Water, Steam, and Radioactive Waste Containing Components of Nuclear Power Plants,”

- RG 1.28, “Quality Assurance Program Criteria (Design and Construction),”

- RG 1.54, “Service Level I, II, III and In-Scope License Renewal Protective Coatings Applied to Nuclear Power Plants,”

- RG 1.57, “Design Limits and Loading Combinations for Metal Primary Reactor Containment System Components,”

- RG 1.61, “Damping Values For Seismic Design Of Nuclear Power Plants,”

- RG 1.136, “Design Limits, Loading Combinations, Materials, Construction, and Testing of Concrete Containments,”

- RG 1.160, “Monitoring the Effectiveness of Maintenance at Nuclear Power Plants,”

- RG 1.163, “Performance-Based Containment Leak-Test Program,”

- RG 1.199, “Anchoring Components and Structural Supports in Concrete,”

- RG 1.216,” Containment Structural Integrity Evaluation for Internal Pressure Loadings Above Design-Basis Pressure,”

- RG 1.217, “Guidance for the Assessment of Beyond-Design-Basis Aircraft Impacts,” and

- RG 1.243, “Safety-Related Steel Structures and SC Walls for Other than Reactor Vessels and Containments.”

Applicable Industry Codes and Standards

- American Society of Mechanical Engineers (ASME) B&PV Code, Section III, Division 1 and Division 2, and Section XI [11]

- American National Standards Institute (ANSI)/American Institute of Steel Construction (AISC) N690-18, Appendix N9 [12]

- American Society of Civil Engineers (ASCE) 4-16

- Structural Engineering Institute (SEI)/ASCE 11

- American Concrete Institute (ACI) 349.3R

Technical Evaluation

Description of BWRX-300 Reactor Building and Containment Structures

As stated in the LTR, the BWRX-300 integrated RB consists of the RB which encloses the containment structure and the SCCV containment structure. The integrated RB is constructed using diaphragm plate (DP) SC (DP-SC) modules and is deeply embedded so that most of these related structures are below grade. The RB structure is a cylindrical-shaped, shear wall building. The walls, floors, roof, and mat foundation are primarily constructed using DP-SC modules. The SCCV consists of a cylindrical wall, mat foundation, and top slab also constructed using DP-SC modules. The RB, containment, and containment internal structures are integrated at the DP-SC mat foundation, wing walls, and floor slabs, which includes the pool slab and walls.

Description of Steel-Plate Composite (SC) Structures

As described in LTR Section 3.4, "Steel-Plate Composite Structures," DP-SC structural modules are constructed by placing concrete between two steel faceplates that serve as main reinforcement and permanent formwork. Steel ties, in the form of continuous diaphragm plates with holes to allow concrete flow, and steel headed stud anchors, are used to develop composite action between the concrete and the steel faceplates and to maintain strain compatibility between concrete and steel. The headed stud anchors are used in addition to ties to anchor the faceplates to the concrete infill and control faceplate local buckling. These DP-SC modules differ from traditional SC modules due to the configuration of the steel ties used to connect the two faceplates to provide composite action, serve as out-of-plane shear reinforcement, and prevent delamination. Further, traditional SC modules use discrete tie bars of round or rectangular cross-section, whereas DP-SC modules use continuous diaphragm plates with holes.

Codes & Standards Evaluation

Current design codes in Canada and the United States do not address the use of SC structural systems as a containment pressure boundary and do not address the use of DP-SC for any structure including containment. Therefore, GEH proposed specific design rules for the SCCV and RB structures by adapting the content structure and provisions as well as proposing modified criteria of the below codes and standards:

Containment Vessel (SCCV):

- Canadian Codes & Standards: CSA N287 series [7] are for concrete containment structures and do not clearly provide requirements for DP-SC containment structure. They do, however, provide for the general performance requirements of containment structures and are in this way applied to the CNSC's review on this topic.

- The 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, to the extent applicable and supplement with additional technical bases.

Reactor Building:

- Canadian Codes & Standards: CSA N291 [8] is the design code in Canada that would be applied to RB, but it and its cascading references CSA A23 series (for concrete design) and CSA S16 (for steel design) do not include provisions to cover the proposed DP-SC structure.

- U.S. Codes & Standards: American National Standards Institute (ANSI)/American Institute of Steel Construction (AISC) N690-18, Chapters NM, NN, and Appendix N9 [12], American Society of Civil Engineers (ASCE)/Structural Engineering Institute (SEI) 43-19 [13]

The LTR describes the approach and technical basis for meeting applicable regulatory requirements by adapting, to the extent applicable, cognizant existing codes and standards which are further supplemented and modified by design rules that are specific to the BWRX-300 SC structures. GEH, in its LTR, requested the CNSC and NRC design-specific approval for the use of these codes and standards.

Prototype Test Observation

At the invitation of GEH, the NRC and CNSC staff observed one of 15 scaled prototype tests performed as part of the U.S. Department of Energy sponsored National Reactor Innovation Center (NRIC) Demonstration Program Phase 1 (Detailed Design and Structural Performance Testing) at Purdue University. The main objectives of the prototype tests are to demonstrate the structural performance of SteelBricks™ under design-basis and beyond design-basis loading conditions. The scaled prototype specimens were constructed and tested for various loading conditions applicable for containment (i.e., pressure-retaining) and non-containment applications. The specific test observed by the NRC and CNSC staff was the 1:3 scale combined In-Plane (IP) Shear + Out-Of-Plane (OOP) Shear test performed on April 7, 2023.

The test was conducted by applying constant force in the OOP direction and then applying gradually increasing/decreasing cyclic forces in the IP direction depicting the design demands of lateral earth pressures and cyclic design-basis earthquake (DBE) loads, respectively. The scaled test specimen was designed to be representative of the RB exterior wall-to-mat foundation connection. It included a splice connection at the interface between the mat foundation module and the wall module. During the tests, the displacements and strain measurements at the critical locations were viewed live and recorded for future engineering assessments.

The structural integrity of the sections of SteelBricks™ were maintained until to the end of the tests where IP was loaded to three times the yield displacement. The test was terminated before failure of any connection members. Therefore, the test verified that the connection details can withstand design demands without connection failure at SteelBricks™ RB-to-basemat connections.

Code equations in AISC N690 were initially applied in developing models to demonstrate how the specimen will behave. The test results passed the acceptance criterion by exceeding the expected IP flexural capacity, which was based on the linear interaction between IP and OOP flexure capacities. The failure mode was also observed to be dominated by ductile steel behavior. The CNSC and NRC staff were able to witness how the test results confirmed that the applicable code equations in AISC N690 (as-is or modified where necessary) can conservatively estimate the SC performance under DBE and lateral earth pressures.

The NRC and CNSC staff also toured the Purdue Bowen Laboratory, observed test set-up, and examined tested specimens from previously conducted IP shear, OOP shear and missile impact tests.

Since the SteelBricks™ testing, additional NRIC testing has been conducted on the DP-SC modules that will be used in the BWRX-300, in testing Phases 2 and 3. At the time of this report, CNSC had only received the results of the Phase 1 testing and therefore cannot comment on whether the Phase 2 and 3 testing demonstrated that the DP-SC modules that will be used in the BWRX-300 meet requirements as provided above, or on the testing methodology used in these tests.

Technical Review Results

The review process followed the steps specified in the NRC LTR review process as per LIC 500, "Topical Report Process". During the audit phase of the review, NRC and CNSC staff raised technical questions, which were addressed by GEH in a timely and satisfactory manner. After the audit concluded, each regulator used the results to feed into their respective process. CNSC staff sent comments to OPG and NRC staff issued RAIs to GEH. Responses have been provided to both comments and RAIs. The responses provided further information on the questions and concerns related to the LTR. In response to the audit questions and subsequent RAIs and comments, GEH prepared and issued revision 2 of the LTR, implementing changes to address regulatory concerns. At the time of this report, CNSC staff were working with OPG on addressing further application of the LTR in the LTC process and development of the design. The NRC staff completed its review of LTR revision 2, resulting in the issuance of an SER.

Some of the comments and limitations and conditions raised by regulatory staff were on the following subjects:

- Design information (drawings, specifications and calculations)
- Limitation on maximum DP-SC section thickness
- Limitation on maximum concrete compressive strength
- Limitation on the maximum design temperature
- Design methodology of connections between DP-SC elements (e.g. DP-SC wall element and DP-SC RB roof, DP-SC wall-to-slab connections, splices of DP-SC modules, etc.)
- Consideration of residual stresses
- Corrosion protection
- Examination techniques for inspection and testing (in addition to visual examination)
- Preservice and inservice inspection and testing
- Dissimilar metal interaction
- Constructability verification

Further information on the identified subjects is expected to be provided to regulatory staff when design development progresses.

CNSC–NRC Joint Conclusion

The CNSC and the NRC have reviewed the LTR and conclude that GEH's LTR on the design of BWRX-300 SCCV and RB provides a reasonable approach for demonstrating the acceptability of the design methodology and preparing additional submittals regarding the BWRX-300 SMR design and licensing basis related to steel-plate composite structures. GEH recognizes the proposed design is not covered in its entirety by the existing design codes and has an appropriate approach on bridging those gaps. The following key points have been identified during the review:

- The overall licensing approach for BWRX-300 steel plate composite structures is acceptable. Acceptability of information provided in subsequent submissions is important and will be duly evaluated when available.

- The development of the design-basis was incomplete as further testing was outstanding at the time this report began development. This testing is being done as part of a U.S. Department of Energy sponsored NRIC Advanced Construction Technology project in the U.S. The LTR provides more information on the design and its technical basis. Additional testing information is expected to be provided in future licensing submissions.

- There are many design details that both regulators expect to be addressed beyond this topical report review, as outlined in section 4.5. Examples of those are SC basemat thickness, typical steel-SC materials dimensions, material specification, and dissimilar metal interactions on the inner side of the SC spent fuel pool structure, including the details of the process for connecting stainless steel to carbon steel, damping, ductility, constructability and aging management considerations.

- The Canadian Regulatory Framework requires that aging management is considered as part of design and this needs to be demonstrated for the SC design.

References

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4. GE-Hitachi Nuclear Energy Americas LLC (GEH) White Paper – "NEDO-33988, Revision 0, BWRX-300 Steel-Plate Composite (SC) Containment Vessel (SCCV) and Reactor Building Structural Design White Paper," dated October 14, 2022 (ML22287A177, e-doc # 6891712)

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18. CNSC REGDOC 2.6.3, “Fitness for service Ageing Management” 2014

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20. NRC Staff Request for Additional Information Letter No. 18 for Topical Report NEDC-33926P (ML24033A206)

21. Final Safety Evaluation of GEH Topical Report NEDO-33926/NEDC-33926P, Revision 2 (Public) (ML24220A017)

22. NEDO-33926, Revision 2, “BWRX-300 Steel-Plate Composite Containment Vessel (SCCV) and Reactor Building (RB) Structural Building” (ML24110A134)

23. NRC-CNSC Joint Report on BWRX-300 Steel-Plate Composite Containment Vessel and Reactor Building Structural Design White Paper (ML23100A032)