Joint Report on GE Hitachi's Containment Evaluation Method

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

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Acronyms

ABWR	advanced boiling water reactor
ART-SMR	Advanced Reactor Technologies – Small Modular Reactor
ATWS	anticipated transient without scram
BWR	boiling water reactor
CFR	Code of Federal Regulations
CAMP	Code Application and Maintenance Program
CANDU	CANada Deuterium Uranium
CNSC	Canadian Nuclear Safety Commission
CSA	Canadian Standards Association
EMDAP	evaluation model development and assessment process
ESBWR	economic simplified boiling water reactor
GEH	General Electric Hitachi
IC	isolation condenser
ICS	isolation condenser system
LOCA	loss-of-coolant accident
LTR	licensing topical report
MOC	memorandum of cooperation
MOU	memorandum of understanding
NCG	noncondensable gases
NRC	Nuclear Regulatory Commission
OPG	Ontario Power Generation
PCCS	passive containment cooling system
PIRT	phenomena identification and ranking table
RCS	reactor coolant system
RG	Regulatory Guide
RPV	reactor pressure vessel
SE	safety evaluation
SMR	small modular reactor
SRP	standard review plan
TVA	Tennessee Valley Authority
VDR	vendor design review

1 Introduction

This report documents the joint review activities between the United States Nuclear Regulatory Commission (NRC) and the Canadian Nuclear Safety Commission (CNSC) to assess a method for predicting the conditions inside the containment vessel following a loss-of-coolant accident (LOCA) for the GE-Hitachi Nuclear Energy Americas, LLC (GEH) BWRX-300 reactor design. The method has been proposed by GEH in the form of a licensing topical report (LTR) entitled *"BWRX-300 Containment Evaluation Method"* [1] (hereafter referred to as the LTR). This LTR supports future licensing activities for GEH's BWRX-300 reactor design.

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*, its associated regulations or the *Canadian Nuclear Safety Commission 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 licence under Section 24 of the *Nuclear Safety and Control Act* or under Section 103 of the Atomic Energy Act of 1954. The conclusions in this collaborative report are of the CNSC and NRC staff.

The NRC conducted its technical review of the LTR and CNSC evaluated the same technical information. Both regulators evaluated how the information would be used in each country's regulatory infrastructure and confirmed items that could be documented for mutual understanding. This work is performed under the framework of a memorandum of understanding (MOU) between Canada and the United States as described in Section 1.1.

1.1 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 memorandum of cooperation (MOC) [3] aimed at enhancing technical reviews of advanced reactor and small modular reactor (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 SMRs and advanced reactor technologies in the future. Cooperative activities are conducted with acknowledgment of differences in the regulatory frameworks and licensing processes 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) Sub Committee. This sub committee approves and prioritizes work plans to accomplish specific cooperative activities under the MOC. The sub committee functions according to the terms of reference established in 2020 [4].

The ART-SMR Sub Committee has overseen several cooperative technical reviews between staff at the CNSC and NRC. A listing of the topics covered to date include, but are not limited to:

- reactor pressure vessel (RPV) construction codes pertaining to the X-Energy high temperature gas reactor [5]
- benefits of the risk-informed, performance based regulatory approach (described by NEI 18-04) [6]
- identification of initiating events pertaining to the Terrestrial Energy integral molten salt reactor

This report documents the cooperative review activities performed on GEH's methodology for evaluating the robustness of containment in response to LOCAs for the BWRX-300 reactor design.

1.2 Background on GEH Engagement with the Regulators

An overview of the key dates in the execution of this joint review is provided in Figure 1 below.

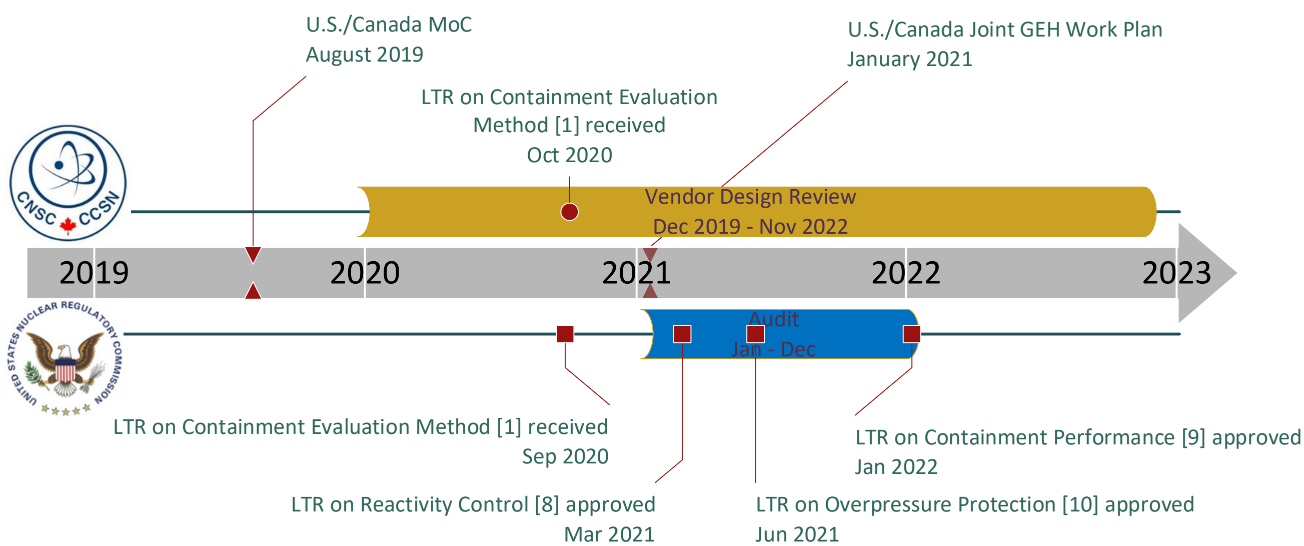

Figure 1: Timeline of GEH Engagement with the CNSC and the NRC

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 construction permit application. To date, GEH has submitted five topical reports to the NRC for review. The topical report process is used by the NRC and the nuclear industry in the United States to improve the efficiency of other licensing processes by allowing NRC staff to review proposed methodologies, designs, operational requirements, or other safety-related subjects on a generic basis. Once found to be acceptable for use and approved by NRC staff, licensing topical reports may be implemented by reference in licensing applications by multiple applicants or licensees. This minimizes industry and NRC time and effort by providing a streamlined review of a subject with applicability to more than one facility.

Four of these LTRs have been focused on the area of reactor systems and design,

- *"BWRX-300 Reactivity Control"* [8],
- *"BWRX-300 Containment Performance"* [9],
- *"BWRX-300 Reactor Pressure Vessel Isolation and Overpressure Protection"* [10]; and,
- *"BWRX-300 Containment Evaluation Method"* [1] (the subject of this joint review).

The NRC staff has completed its review and issued safety evaluations (SEs) approving the first three LTRs listed above. In these approvals, the staff concluded that GEH's novel approach to mitigating the effects of the large break LOCA and anticipated transient without scram (ATWS) events were acceptable in concept and subject to confirmation during future licensing activities. Approval of these initial LTRs focusing on specific areas of regulatory risk provided GEH the regulatory predictability needed to proceed with the BWRX-300 reactor design, leading to the submittal of the containment evaluation method LTR.

GEH submitted the subject LTR for NRC review as Revision 0 in September 2020. Over the course of the joint regulatory review, GEH revised the LTR twice in response to NRC requests for additional information, with Revision 2 submitted in December 2021 [1]. The LTR documents GEH's methodology for calculating the peak containment pressure for both large break LOCA scenarios and small break LOCA scenarios. NRC accepted this topical report for review in November 2020 and issued a SE [11] in February 2022. As part of its review, NRC staff conducted a regulatory audit from January 2021 to December 2021, and documented the results in an audit report [12]. Additionally, NRC staff performed sensitivity and independent confirmatory analyses using the TRACG, GOTHIC, TRACE and MELCOR computer codes. The staff used these analyses to focus the review on the most safety significant issues, to aid in the staff's understanding, and to resolve the issues. Additional detail on these analyses is provided in the NRC staff's SE. The SE states the detailed bases for the NRC conclusions regarding the LTR set forth in this report.

GEH Engagement with the CNSC

GEH entered into a service agreement with the CNSC in December 2019 [13] to participate in a pre-licensing review called a vendor design review (VDR). A VDR is an optional service that the CNSC provides to assess a vendor's reactor design readiness. The primary purpose of a VDR is to offer feedback to the vendor about how it is addressing Canadian regulatory requirements and CNSC expectations in its design and design activities. A VDR is organized as per REGDOC-3.5.4 [14] into 19 topics called focus areas. More details on Canada's pre-licensing engagement strategy are available in Section 2.1.1 of Reference [6].

Over the course of the VDR, which spans approximately three years, GEH will release technical information in staged packages, host technical meetings for CNSC staff, and respond to requests for further information. At the time of issuing this document, GEH has submitted all four information packages to the CNSC for review. The culmination of the CNSC staff review is documented in an integrated report that identifies technical clarifications, findings, and any fundamental barriers to licensing.

The LTR [1] was received under focus area 8 on containment under the VDR in October 2020, as shown in the timeline of Figure 1. The insights gained from this joint review of the LTR will also be of benefit to focus area 7 on the emergency heat removal system as well as focus area 10 on safety analysis. Insights presented in this joint review will be carried forward, as appropriate, into the VDR integrated report scheduled for completion in November 2022.

2 Scope and Objectives for this Joint Review

The work plan [7] issued at the project onset in January 2021, describes the scope of work and objectives as follows:

> "To share regulatory experiences and insights for the Boiling Water Reactor X-300 (BWRX-300) small modular reactor design. Specifically, the scope of work is to perform a collaborative review of the licensing topical report titled BWRX-300 Containment Evaluation Method [1]. An exchange of information between the CNSC and U.S. NRC will cover safety review methodologies, regulatory approaches, and treatment of unique aspects of the *BWRX-300 Containment Evaluation Method*."

The review scope is focused on the methodology and not necessarily the design inputs or simulation outputs. It is understood that the BWRX-300 reactor design is currently still under development and design inputs may and are likely to change as the design matures. While the design inputs (e.g., water volumes, reactor dimensions) are reasonable approximations of known parameters for the purpose of trialing the methodology, precision is not critical. Likewise, the outputs from the code simulations are considered sample results intended to demonstrate an ability of the computer codes to produce key figures of merit that will be evaluated in future licensing applications against regulatory acceptance criteria.

3 Use of Computer Codes in Safety Analysis

This section provides a summary of the regulatory requirements and guidance used by each regulator along with a discussion of the underlying safety principles and objectives important in the use of the requirements and guidance.

3.1 CNSC Regulatory Framework

New reactor designs are assessed in Canada against design, analysis and programmatic requirements documented in a suite of regulatory documents under the *Nuclear Safety and Control Act*. Most relevant to new builds is REGDOC-2.5.2, "*Design of Reactor Facilities: Nuclear Power Plants"* [15]. Licence applicants for new nuclear power plants must demonstrate compliance with REGDOC-2.5.2. REGDOC-2.5.2 is also used to evaluate designs and supporting processes, earlier in the design development stages as part of pre-licensing VDRs.

The LTR pertains to the analysis method proposed by GEH to study the effects of a LOCA on the primary containment vessel. REGDOC-2.5.2 covers safety analysis of this type through reference to REGDOC-2.4.1, "*Deterministic Safety Analysis"* [16]. REGDOC-2.4.1, Section 4.4.5, gives the following requirement related to the use of computer codes in safety analysis:

> "Computer codes used in the safety analysis shall be developed, validated, and used in accordance with a quality assurance program that meets the requirements of CSA N286.7-99, *Quality Assurance of Analytical, Scientific, and Design Computer Programs for Nuclear Power Plants"*.

Computer codes are required to meet the Canadian Standards Association (CSA) N286.7-99 [17] that is referenced in REGDOC-2.5.2 and REGDOC-2.4.1. Of particular importance in N286.7 is Section 10.1 on "Requirements for Use" which specifies how computer programs shall be validated for their intended use. Validation is the comparison of the results of the computer program with measurements, experimental data, or known analytical or numerical solutions, so that the accuracy or uncertainty of a particular application can be determined. Section 9 of N286.7 provides examples of validation methods which an organization can undertake by comparing the computer program results to:

a) experimental data
b) commissioning data and operating experience
c) solutions to hand calculations
d) solutions to standard or benchmark problems
e) closed mathematical solutions
f) results of another validated computer program

N286.7 requires that the validation results be reproducible, performed by qualified persons and documented in a validation report. For the CNSC, much effort was focused on reviewing the validation basis of TRACG given that this proprietary computer code used by the vendor is new to the CNSC.

In addition, GEH will be expected to identify any validation effort that was performed for the GOTHIC code to close any gaps between BWRX-300 and the application of GOTHIC to the Economic Simplified Boiling Water Reactor (ESBWR) design.

Further computer code requirements listed in Section 10.1 of N286.7 are under review:

> Each organization shall ensure the proper use of computer programs by requiring that:
>
> (a) computer programs are validated for the intended use (covered in the two preceding paragraphs)
>
> (b) only those physical states are analyzed that are within the documented range of the computer program's applicability
>
> (c) the input data is verified to ensure that it adequately represents the physical system or process analyzed
>
> (d) the derivations and sources of input data are documented in a form that facilitates independent review
>
> (e) the configuration of the computer program and the input data are identified so that results can be reproduced
>
> (f) the results produced by the computer program are reviewed to confirm that they are reasonable, and
>
> (g) user qualifications are specified and the necessary training is provided to minimize the effect of user dependency

REGDOC-2.4.1 provides guidance in the subsections that follow on computer code applicability, code validation, quantification of accuracy, and physical representations.

3.2 NRC Regulatory Framework

New reactor designs in the United States are assessed against the criteria outlined in Title 10 of the Code of Federal Regulations (10 CFR) Part 50 *"Domestic Licensing of Production and Utilization Facilities,"* or 10 CFR Part 52 *"Licenses, Certifications, and Approvals for Nuclear Power Plants,"* although the substantive regulations are largely equivalent. Additionally, the NRC is currently developing another licensing pathway for new reactor designs, which is tentatively styled as 10 CFR Part 53 *"Risk Informed, Technology-Inclusive Regulatory Framework for Advanced Reactors."* The Part 53 regulations are currently in development and have not yet been published as a proposed rule. Further details on the NRC's licensing process for each of these separate pathways is available on the NRC's public website.

GEH, as part of a joint effort with the Tennessee Valley Authority (TVA), indicated that the prospective applicant plans to file a construction permit applicant under 10 CFR Part 50 for 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. This includes the initial topical reports that GEH has submitted to the NRC for approval in preparation for submitting a complete reactor design, as discussed in Section 1.3.

GEH has received approval of the *"BWRX-300 Containment Performance"* topical report, which provides the design requirements, acceptance criteria, and regulatory bases for the BWRX-300 containment performance design functions. Complete detail of the staff's review can be found in the SE [9]. The review of the subject LTR [1] therefore, focused on the ability of the containment evaluation method to appropriately model the containment design and predict whether the design will meet the previously approved acceptance criteria defined in the containment performance topical report.

The NRC has developed a Standard Review Plan (SRP), NUREG-0800, [18] to assist staff in performing its reviews. SRP Section 6.2.1 *"Containment Functional Design"* [19] and Section 15.0.2 *"Review of Transient and Accident Analysis Methods"* [20] were used by the staff to perform the review of the LTR [1]. These sections of the SRP describe the review process and acceptance criteria for analytical models and computer codes used to analyze accident and transient behavior. The purpose of the review is to verify that the evaluation model is adequate to simulate the accident under consideration. Specifically, the acceptance criteria contained in the SRP are related to the evaluation model, accident scenario identification process, code assessment, uncertainty analysis, and quality assurance plan.

Additionally, the NRC has developed guidance for industry in Regulatory Guide (RG) 1.203 *"Transient and Accident Analysis Methods"* [21] that describes a process that the staff considers acceptable for use in developing and assessing evaluation models used to analyze transient and accident behavior. This Regulatory Guide outlines the basic principles of evaluation model development and assessment:

- determine requirements for the evaluation model
- develop an assessment base consistent with the determined requirements
- develop the evaluation model
- assess the adequacy of the evaluation model
- follow an appropriate quality assurance protocol during the Evaluation Model Development and Assessment Process (EMDAP)
- provide comprehensive, accurate, up-to-date documentation

4 Technical Evaluation

The technical evaluation is organized into sections with text corresponding to the section heading followed by regulatory commentary that is distinguished by grey shading. This commentary provides feedback on the approach taken by GEH and reflects joint CNSC and NRC staff views. Any feedback that has implications on future licensing engagements and/or activities are identified and listed in Section 5 as a "licensing consideration". This term has been introduced in this joint report to set it apart and avoid confusion with established regulatory terminology between the two nations.

4.1 Description of the BWRX-300 Reactor and Containment Design

A brief overview of the reactor design is provided in this section prior to describing the analytical approaches. The details that follow describes the design features most relevant to the analysis under examination. For more information on the nuclear power plant design, refer to the Advanced Reactor Information System database on the International Atomic Energy Agency website.

The BWRX-300 is a boiling water reactor (BWR) technology. It operates in the thermal neutron spectrum using light water as the reactor coolant and moderator. The reactor is sized to generate 300 MW_e at around 870 MW_{th}. The BWRX-300 is the 10^{th} design iteration in the line of reactor designs by GEH. The BWRX-300 builds upon the passive design features of its larger capacity predecessors; the Advanced BWR (ABWR) and the ESBWR. Flow through the reactor is accomplished by natural circulation without any recirculation pumps.

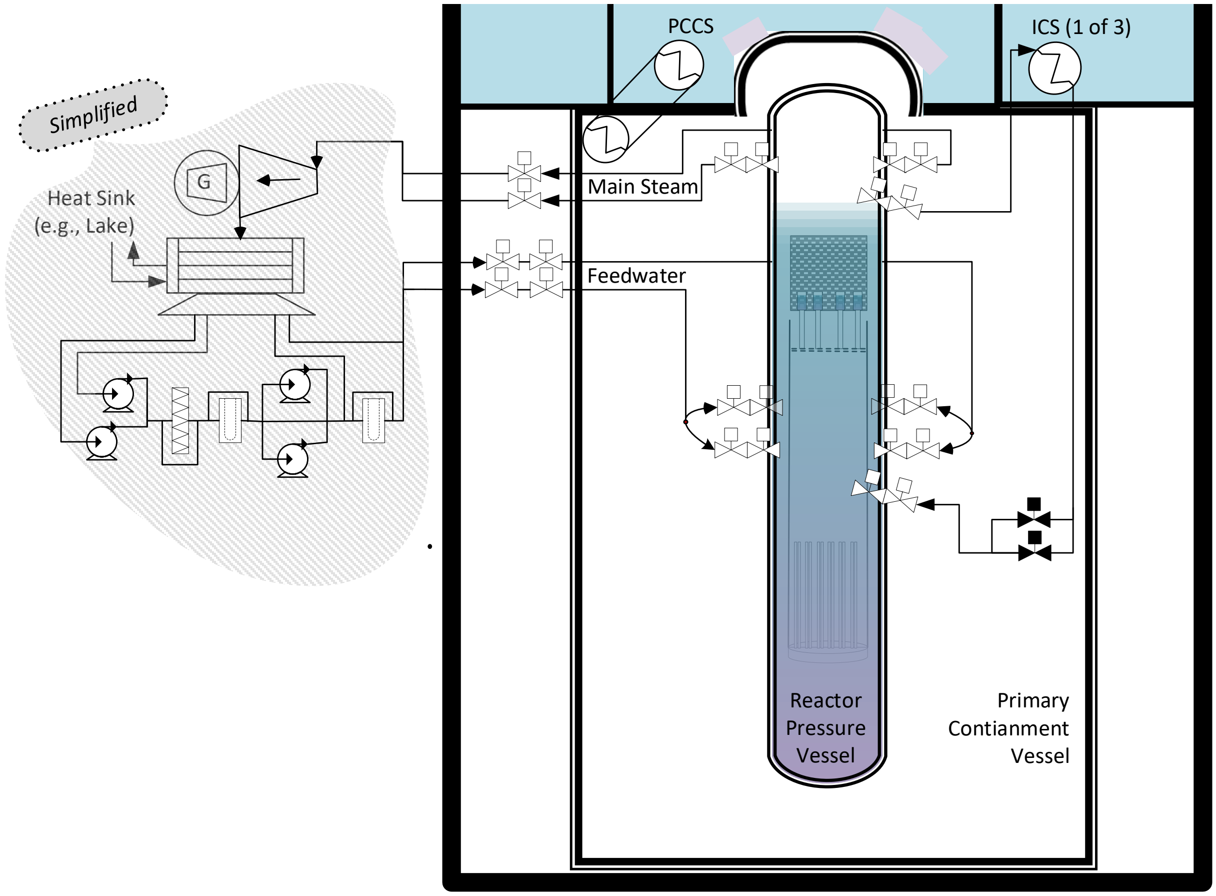

Figure 2 provides a simplified process flow diagram of the BWRX-300. Key nuclear system attributes and functions are provided below:

- the containment is a dry enclosure, near atmospheric pressure during normal operation
- the containment is filled with nitrogen during normal operation
- the containment includes a passive containment cooling system (PCCS) that is always in service without any reliance on actuation signals, operator intervention, or active components to function
- there are three independent isolation condenser system (ICS) trains that remove decay heat following reactor shutdown with each train rated for 33 MW_{th}
- when a demand signal is received by the ICS, the RPV isolation valves close and the isolation condenser liquid return lines open to place the system in service.

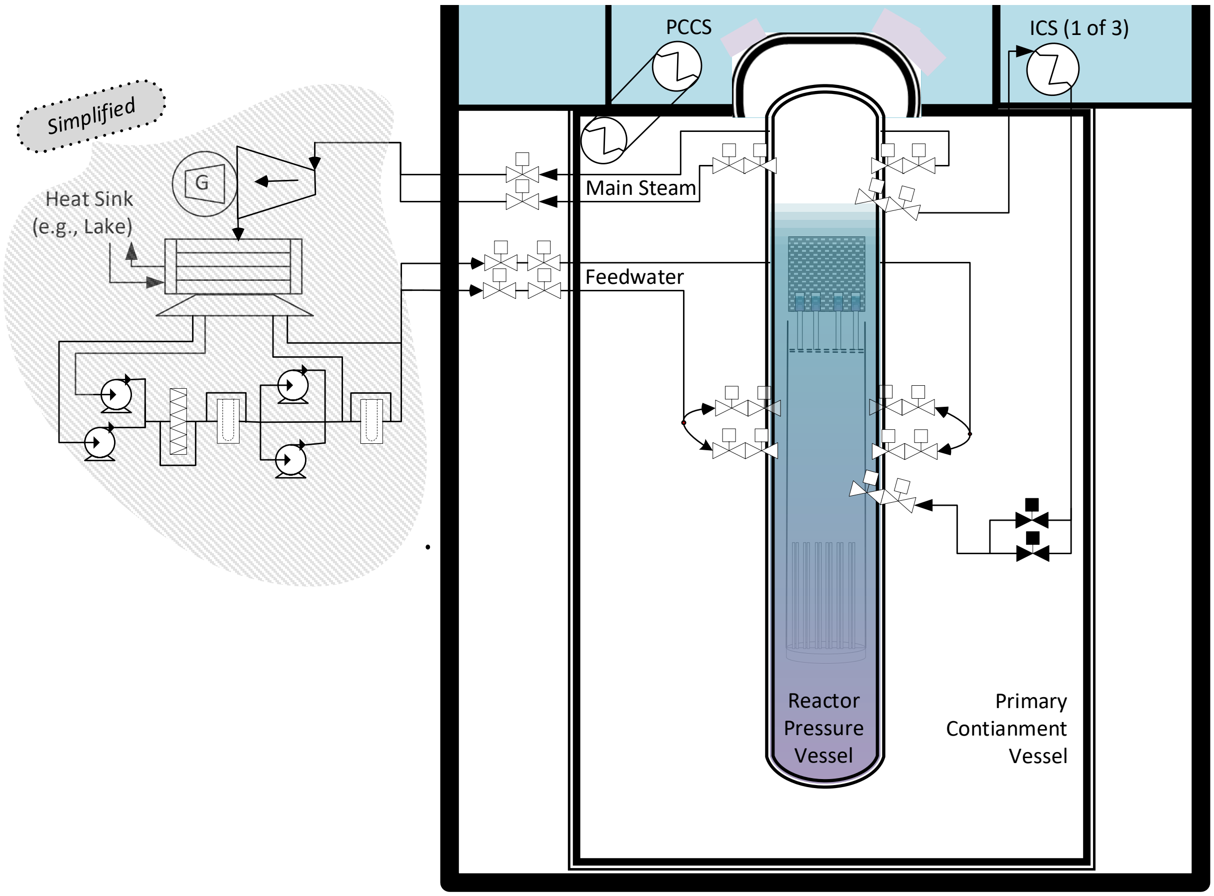

Figure 2: Simplified Process Flow Diagram of the BWRX-300

4.2 Selection of Accident Conditions to Challenge Containment

To understand the necessary performance of the containment vessel, GEH is proposing a method of analysis to predict the containment conditions following the demands of a postulated accident. By way of examining the set of design basis accidents, GEH elected to evaluate the challenges to fuel and containment integrity following LOCAs discharging high energy reactor coolant into containment. As part of the LTR, GEH identifies what pipes to simulate for rupture (e.g., main steam lines, feedwater lines, instrument lines), the specific location in the piping segments where such ruptures occur, and the conditions within those pipes to maximize, or bound, the pressure and temperature stresses transferred to the containment vessel.

The LTR classifies the postulated pipe breaks into large and small because automatic safety functions have a significant effect on the accident progression depending upon what piping ruptures. All large pipes (i.e., {{ }}) are equipped with isolation valves mounted directly to the RPV, which close upon detection of a break inside containment. This differs for the small break classification in that the discharge continues without mitigation because no such isolation valves are present.

During the course of operation, the reactor will experience varying plant operating states (e.g., reactor power level, water levels, feedwater temperature). These operating parameters describe the steady-state nuclear and hydraulic conditions and are prerequisite initial conditions for which the analysis execution depend. It is an intractable problem to study all possible plant configurations. Instead, assumptions are made to effectively maximize the liquid and steam discharge into containment. As was the approach in selecting the pipe break conditions described above, GEH applied engineering judgement to bound the range of possible variables for the plant operating conditions. GEH opted to study two cases; a base case and a conservative case, in accordance with elements of NRC RG 1.203. The base case uses nominal initial conditions and nominal modelling parameters. The conservative case, as the label suggests, reflects initial conditions that are biased in the conservative direction. The initial conditions for both cases are specified in Table 5-2 of the LTR.

Modeling the response of the plant following the postulated pipe rupture follows a similar conservative approach with assumptions that effectively maximize the challenge to containment. Examples of modelling assumptions include:

- discharge rate is calculated with the containment at atmospheric pressure to maximize the discharge rate
- the reactor trip is due to a loss of offsite power or load rejection
- at least one of the three isolation condensers is out of service

Regulatory Commentary
NRC and CNSC staff determined that the modeling of the initial RPV conditions is acceptable. The NRC and CNSC staff reviewed the initial conditions for the base and conservative cases and determined they were correctly selected and biased, as necessary, to produce appropriately conservative results. Additionally, the NRC staff performed independent sensitivity studies using TRACE that supports this.

Given that event classification of design basis accidents in Canada extends to 10^{-5} per year, CNSC staff will check in future licensing engagements if this criterion has implications in the selection of a bounding accident.

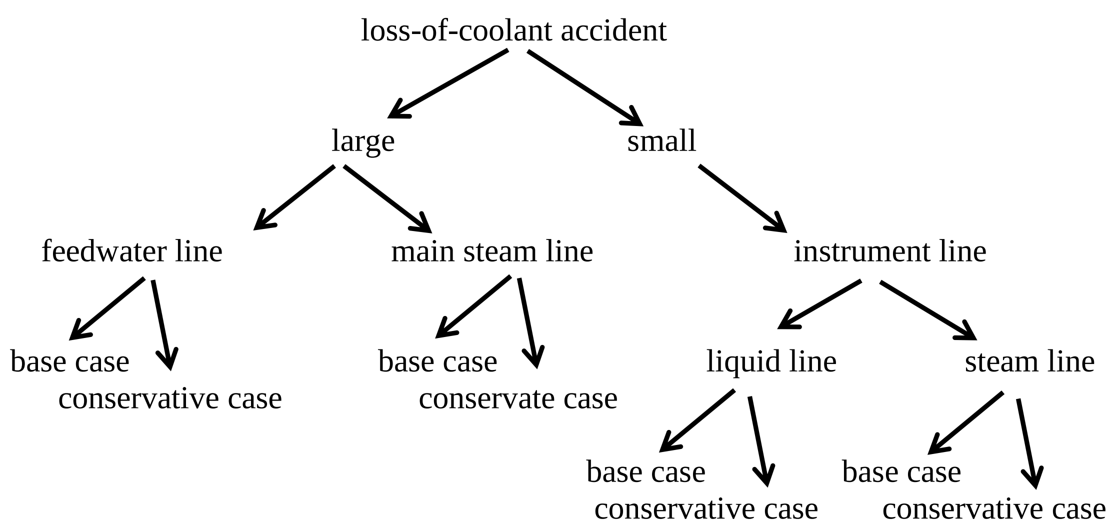

Figure 3: Demonstration Cases

4.3 Selection of Computer Codes

GEH has selected TRACG version 04.02.75.00 and GOTHIC version 8.3 to perform containment analysis for the BWRX-300. In the LTR, mass and energy outputs from TRACG are input as boundary conditions into the GOTHIC model to evaluate the containment response.

TRACG is based on a two-fluid thermal hydraulic model for the reactor vessel, the coolant system, the containment, and a three-dimensional kinetics model for the reactor core. It is considered a best-estimate code and has been used by GEH for several generations of BWRs in design, safety assessment, and licensing. This has resulted in the development of an extensive experimental base.

GEH uses a systematic approach for TRACG qualification, consisting of the following major qualification categories:

1. separate effects tests
2. component performance tests
3. integral system effects tests
4. standard BWR nodalization
5. BWR plant tests

The verification and validation framework developed by GEH for use in the United States is similar to Canadian regulatory expectations and framework.

GOTHIC is a commercially available computer code for modeling multiphase, multicomponent fluid flow that has been developed consistent with U.S. regulatory requirements in 10 CFR Part 50, Appendix B, and meets GEH software quality standards. GOTHIC is considered an industry-standard code for performing both containment design basis accident analyses and analyses to support equipment qualification. The code has a flexible noding structure that allows both lumped parameter and 3-D modeling capabilities; the NRC and CNSC staff have verified that GOTHIC has undergone extensive review and validation against a large array of tests.

The containment analyses in this LTR are performed using the latest GOTHIC version 8.3. Additionally, the LTR states that "Future BWRX-300 containment analyses may be performed using newer versions of the GOTHIC code provided the newer versions meet the same 10 CFR Part 50 Appendix B quality requirements and changes in calculated results for our BWRX-300 containment application caused by any code changes can be successfully dispositioned."

4.3.1 TRACG and GOTHIC Knowledge Base of the Regulators

This section describes the prior experience base the NRC and CNSC have with the TRACG and GOTHIC computer codes for context.

<u>TRACG Knowledge Base</u>

The NRC has extensive experience with the TRACG code. The NRC staff has reviewed and approved the application of TRACG for both the ESBWR and operating BWRs. The NRC staff was initially introduced to the TRACG code in its design certification review of the GE Simplified BWR. NRC staff also conducted extensive regulatory review on the design certification for the ESBWR, in which GEH applied the TRACG code to the ESBWR LOCA and containment analysis, as documented in topical reports [22][23][24][25]. NRC staff also evaluated use of the TRACG code during its review of the ESBWR design certification, as described in the staff's SE report [26].

In contrast, the CNSC does not have any prior experience with the TRACG code. CNSC staff has experience with thermal hydraulic codes for safety assessments of CANDU reactors, such as CATHENA, which was developed by Canadian Nuclear Laboratories, and TUF, which was developed by Ontario Power Generation (OPG). Both of these codes are the recognized industry standard toolset codes used in Canada. A computer code is recognized as an industry standard toolset if it complies with a set of quality assurance standards for plant simulations. The CNSC is a member of the Code Applications and Maintenance Program (CAMP), which is a user's group for NRC codes. As such, the CNSC has access and training to TRACE and RELAP5, which are similar to TRACG.

<u>GOTHIC Knowledge Base</u>
The NRC has extensive experience with the GOTHIC code. The staff has reviewed and approved its use for containment analyses for many operating reactors and new reactor designs such as the APR-1400. GOTHIC can be used for a wide variety of plant operations to support analyses involving single and multiphase heat transfer and fluid flow, provided that the application of the code is consistent with the underlying physical basis and assumptions as well as the code validation basis. Typical applications of GOTHIC include evaluation of containment response to the full spectrum of energy line breaks within the design basis envelope. Applications of GOTHIC may include, but are not limited to, pressure and temperature determination, equipment qualification profiles, inadvertent system initiation, and degradation or failure of engineered safety features.

In Canada, GOTHIC is recognized as the containment and severe accident industry standard toolset code. GOTHIC has been relied upon by past and active operating CANDU licence holders to support deterministic safety analyses. The CNSC staff therefore has experience with the computer code GOTHIC for modelling a variety of CANDU nuclear power plant containment applications. Such experience with GOTHIC includes the prediction of hydrogen mixing and distribution.

4.4 TRACG Modelling

The LTR describes the use of TRACG to predict mass and energy releases to containment. The regulatory review focused on both changes made to the code that postdate prior approvals and the application of the method to the BWRX-300 design.

4.4.1 Physical Phenomena and System Response Behaviour of Interest

The NRC and CNSC staff reviewed the various phenomena modeled in the method and considered whether they were modeled appropriately. GEH did not perform a specific phenomena identification and ranking table (PIRT) exercise for the BWRX-300 TRACG mass and energy method per NRC RG 1.203. Instead GEH relies on previous evaluations done for the ESBWR to determine the important phenomena and the appropriate uncertainties to be applied.

Regulatory Commentary
Because the mass and energy release modeled in this LTR falls within that in the ESBWR qualification range, the NRC staff determined that use of TRACG is appropriate.

<u>Decay Heat</u>
GEH modeled the core via a 3D vessel component using a standard-length generic Global Nuclear Fuel bundle type. {{

}}. GEH uses ANS 5.1-1979 nominal as the decay heat curve to compute decay heat power for the base cases, and for the conservative cases ANS 5.1-1979 plus 2 sigma is used, consistent with previous precedent in the ESBWR approval.

<u>Critical Flow</u>
The BWRX-300 incorporates flow limiters in the main steam lines. These flow limiters reduce the flow area in the main steam lines. {{

}}.

Regulatory Commentary
The NRC and CNSC staff have independently assessed the GEH critical flow model using two independent models and determined that it predicts flow consistent with the results of the NRC and CNSC staff models, and, therefore, is appropriate for use.

<u>Isolation Condenser</u>
The LTR states that the isolation condensers are a similar design to those used for the ESBWR. However, the function and connections to the reactor are quite different. The isolation condenser model axial noding used for the BWRX-300 is more detailed than that used for the ESBWR, although physically the size of the component is the same. {{

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Nodalization of some flow components with large volumes representing the ICS pool presents the potential for numerical issues after the peak containment pressure is reached. The main concern of such nodalizations is the potential for numerical problems in a process of solutions of the TRACG main modeling system matrix. Therefore, a possibility of potential oscillations should be considered in important flow components and especially passive systems. The ICS pool side has a nodalization scheme that consists of very large nodes.

Regulatory Commentary
TRACG predicts flow oscillations in the ICS pool after the peak containment pressure is reached during a large break LOCA. TRACG also predicts that these flow oscillations do not have an impact on the peak containment pressure analyses. However, the oscillations should be understood and explained for other transient analyses.

Radiolytic Gases
In BWR reactor systems such as the BWRX-300, radiolytic gases (H_2 and O_2) are generated in the liquid water in the core and are then liberated during boiling. They become mixed in and carried with the steam into the isolation condenser tubing as the isolation condensers begin operation. During the condensation process, these noncondensable gases (NCGs) begin to accumulate in the lower drums and tubing. This buildup of NCGs in the tubing can significantly degrade the isolation condenser ability to condense steam, reducing its heat transfer capacity to cool the reactor. This degradation is critically important for small break LOCA cases where the break is un-isolated {{ }}. The isolation condenser (IC) heat transfer performance must be adequate to cool and de-pressurize the reactor and to maintain a break flow low enough so that sufficient inventory remains within the reactor core region.

The transport, intrusion, and buildup of NCG into the ICs is not explicitly modeled by GEH in the BWRX-300 model. {{ }}. The formulation used for generation of radiolytic gases is a bounding value used generically for the BWR fleet. GEH models radiolytic gas progression into the ICs for determining an overall percentage accumulation in the ICs, {{ }}. Isolation condenser tube heat transfer degradation is modelled via a bias based on the estimated accumulation of NCGs in the model.

Regulatory Commentary
A limitation and condition is identified in the SE [11] related to the accumulation of NCGs in the isolation condensers and is reiterated in this joint report as Licensing Consideration #1.

4.4.2 Development of the TRACG Model

Nodalization
The LTR provides a description of the TRACG nodalization (Figure 5-3) and the modeling used to simulate the RPV and ICS functional behavior for large and small break LOCA transients. The nodalization of BWRX-300 is based on GEH's experience modeling BWR reactors combined with the specific concept design characteristics of BWRX-300. GEH has refined the modeling of the BWRX-300 in relation to the previously approved ESBWR.

Regulatory Commentary
The NRC and CNSC staff have independently determined that the TRACG nodalization appropriately models the key design features of the plant for determining peak containment pressure.

Modeling Biases
The TRACG code has built-in modeling bias parameters that can be specified in code input to modify the results of code runs. They are placed strategically on important coefficients and correlations to add or remove biases with the objective of adding conservatisms or uncertainty to internally calculated nominal code values. In Table 5-1 of the LTR, the vendor identifies seven important parameters used in the assessment. The biases are developed from assessment

comparisons between separate effects test data and TRACG calculations performed with the best-estimate version of the code and are related to probability density functions.

Regulatory Commentary
The containment evaluation method uses appropriately biased initial power conditions with uncertainty added so that main steam and feedwater flow are increased.

NRC staff performed a sensitivity analysis of the relevant modeling parameters listed in Table 5-1 of the LTR and determined that they were conservatively applied for this mass and energy analysis application.

Based on the limited scope of the LTR and because the mass and energy release modeled in this LTR falls within that in the ESBWR qualification range, the NRC staff determined that use of TRACG is appropriate.

4.4.3 Demonstration of the Method

This section describes the accident progression predicted by the TRACG computer code as modelled by GEH.

<u>Large Break</u>
For the large break LOCA cases (greater than one-inch pipe diameter), {{ }} before the RPV isolation valves close, which prevents further mass and energy release to containment. {{ }}. The main steam case results in the maximum predicted mass and energy releases, and in this case, TRACG predicts that long-term cooling is ample since there is minimum core inventory loss as the reactor cools up to 72 hours. The TRACG main steam pipe break analysis bounds all other steam pipe breaks and this analysis is conservative because it analyzes heat removal with only one isolation condenser being available.

The conservative cases start from a higher steam pressure and reactor power but the progression of events is essentially the same as the base cases for both large steam and feedwater line breaks. Consequently, the conservative cases reach a worse outcome in terms of the figures of merit, so these are the cases the NRC and CNSC staff evaluated to determine acceptability of the method.

Regulatory Commentary
The NRC and CNSC staff determined that the base cases are modeled appropriately. The NRC staff reviewed and audited the application of the method in the conservative cases for the large feedwater and steam line breaks. The NRC staff reviewed the main feedwater event results and determined that the mass entering containment is slightly larger than the steam line break, but the energy released is significantly less, so the feedwater line break is not limiting for mass and energy releases.

The NRC staff determined that the modeling and results of the main steam and feedwater cases are reasonable and acceptable for determining mass and energy releases into containment for the BWRX-300 design. The modeling relies heavily on previous assessments performed for the ESBWR design. The NRC and CNSC staff note that the large break steam and feedwater events are modeled as being isolated so quickly that the design differences between the BWRX-300 and ESBWR will have very minimal impact on the results. Therefore, the NRC and CNSC staff determined that GEH's approach was acceptable.

<u>Small Break</u>
For small break LOCA cases ({{ }}), there are no isolation valves, so TRACG models the mass and energy release as continuing throughout the 72-hour transient used for passive plant designs. Consequently, the RCS inventory loss predicted during small breaks can result in core collapsed water level dropping below the top of active fuel, therefore creating the potential for fuel cladding heat up.

Since the small break steam and liquid lines remain un-isolated (i.e., open and flowing) throughout the 72-hour transient, the predicted consequences of these cases are more complex than the large break LOCA scenarios. Predicted results also depend on whether the ICs provide adequate cooling capacity to depressurize the RPV and maintain core inventory.

Because the small steam and liquid pipe breaks are un-isolated, TRACG predicts that break flow continues throughout the 72-hour transient and the containment is conservatively assumed to remain at atmospheric pressure, which maximizes flow out of the break. {{

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The LTR presents TRAGC predictions of the small steam line break out to 72 hours, consistent with analysis practice for passive plant applications that do not consider operator action or alternating current power to mitigate consequences for up to 3 days. The small liquid line pipe breaks are modeled similarly.

For small liquid line breaks, the predicted RPV downcomer level {{

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Regulatory Commentary
The NRC and CNSC staff reviewed the application of the containment evaluation method in the demonstration cases for small breaks and audited associated information. The NRC and CNSC staff reviewed the TRACG estimate of mass flow generated by the small break cases. NRC staff determined that they were conservative in comparison to NRC staff calculations using the Moody correlation at various points of time throughout each small break transient. Accordingly, the NRC and CNSC staff determined that the modeling of the small break steam and liquid cases is reasonable and acceptable for determining mass and energy releases for the BWRX-300 design.

4.5 GOTHIC Modelling

The LTR describes the GOTHIC application method and the BWRX-300 containment model development. GEH describes the:

- identification of relevant inputs and physical phenomena to the containment response
- description of the input model for the BWRX-300 containment, including nominal inputs, assumptions, and correlations
- description of the base cases and results
- nodalization sensitivity studies for the containment and the PCCS
- identification of the key modeling uncertainties and biases used in the conservative case
- benchmark predictions of test data applicable to BWRX-300 containment design
- demonstration analyses to predict containment response for various break sizes and locations using the conservative model
- demonstration of the capability of the PCCS to reduce the containment pressure
- evaluation of the overall conservatism by comparison of the conservative and base case results
- description of the one-way coupling between the TRACG and GOTHIC calculations

The NRC and CNSC staff's review focused on the conservatism of the overall model to produce an acceptable prediction of the BWRX-300 containment response to the mass and energy release calculated using TRACG.

4.5.1 Physical Phenomena and System Response Behaviour of Interest

GEH performed a PIRT exercise to identify the phenomena important to the analysis of the BWRX-300 containment response. Table 6-2 of the LTR lists and ranks the identified phenomena. This list was developed using previously NRC-approved LTRs for prior evolutions of GEH BWR designs and industry standards.

GEH provided the list of relevant phenomena to a group of independent subject matter experts. The experts reviewed and ranked the phenomena in order of importance to the GOTHIC BWRX-300 containment pressure and temperature analysis. The experts also identified any significant phenomena missing from the list. GEH evaluated the phenomena rankings separately for large break LOCA (short-term) and small break LOCA (long-term) transient evolutions given the differences in system response. For large break LOCA, the momentum and inertial effects resulting from the break flow have a significant contribution to the flow circulation, stratification, and heat transfer in the containment. For small break LOCA, the momentum and inertial effects of the break flow are diminished, and buoyancy is the major contributor to the flow circulation, stratification, and heat transfer. GEH used the specific concentric-pipe PCCS design configuration and placement described in the LTR in determining PCCS secondary side phenomena.

GEH eliminated several low ranking phenomena from consideration including 3D/open volume phase separation, liquid entrainment affecting mass flow, void fraction/liquid and gas density, and potential system interactions.

Regulatory Commentary
The NRC and CNSC staff do not agree with the elimination of "potential system interactions" due to the potential in the BWRX-300 design for reverse flow, i.e., from the containment to the RPV, to occur during the long-term transient evolution introducing nitrogen to the isolation condensers. Therefore, the NRC and CNSC staff developed Licensing Consideration #2 and Licensing Consideration #3 in Section 5 to ensure that this phenomenon will be addressed in future licensing activities.

The phenomena identified related to PCCS secondary-side heat transfer were derived using specific design considerations for the PCCS as described in the LTR. Considering the critical role the PCCS plays in mitigating the long-term containment pressure during an un-isolated small break LOCA as well as the potential for reverse flow from the containment back into the RPV, the NRC and CNSC staff developed Licensing Consideration #4.

The NRC and CNSC staff reviewed the qualitative details of the BWRX-300 containment PIRT methodology and found it to be rigorous and consistent with the industry best practices. Sufficient interaction took place among the experts and GEH for consensus building on the PIRT rankings. Licensing considerations related to the PCCS design and the potential for reverse flow will ensure continued applicability of the PIRT.

<u>Containment Mixing for Combustible Gases</u>
The LTR presents discussion and calculations for radiolytic hydrogen and oxygen generation during BWRX-300 design basis accidents. As the radiolytic gases are mixed in steam or liquid water they migrate within the RPV, from the RPV to containment, and within the containment. These gases, mixed in the steam and liquid, are discharged into the containment along with the break flow. This is in addition to the small amount of oxygen initially present in the containment.

Radiolytic hydrogen and oxygen distributions in the containment are not a concern for large breaks in the BWRX-300. However, radiolytic gases may build up in the containment following un-isolated small breaks over time. The LTR assesses if this build-up could pose a safety concern.

{{

}}. Although the volume fraction of radiolytic gases in the steam is small, these gases may accumulate in the dome region over time.

LTR Figure 6-44 shows that the predicted hydrogen volume fraction in the dome region is higher than the main containment volume as expected, but the difference is not large. The figure also shows that the predicted differences in maximum and average combustible gas volumes in the main containment and containment dome are very small. This indicates that sufficient mixing of combustible gases has been demonstrated and eliminates the concern for deflagration in the sub-compartments.

Regulatory Commentary
The BWRX-300 containment combustible gas control is acceptable because: (1) the maximum concentration of combustible gas in containment following a large break is bounded by the un-isolated small breaks; (2) the calculated combustible gases in containment following un-isolated small breaks are far below the deflagration limits; and (3) sufficient mixing in containment sub-compartments eliminates the concern for deflagration in the containment sub-compartments.

4.5.2 Development of the GOTHIC Model

The LTR presents the relevant geometric and thermal hydraulic features of the GOTHIC model used for the BWRX-300 containment analysis.

Figure 6-1 of the LTR shows a schematic of the four components of the GOTHIC model, representing the main section of the containment, the containment dome region above the refueling bellows, the PCCS, and the reactor cavity pool. The main cylindrical containment section and the hemispherical containment dome are nodalized in GOTHIC by using 3-D volumes, as shown in Figure 6-2 of the LTR. {{

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Flow paths are used to model the intake and exit openings of the PCCS units connected to the reactor cavity pool. These PCCS units are placed {{

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The RPV is represented in the GOTHIC model by a blockage corresponding to the outer dimensions of the RPV insulation. The properties of the thermal conductors that are distributed over the RPV surface cells are set to maximize the heat loads from the RPV and piping, which is conservative. The break mass flow rate and enthalpy, as obtained from the TRACG mass and energy release calculations, are specified as a time-dependent boundary condition to the containment GOTHIC model. The fluid temperature in the piping is assumed to be the same as the RPV fluid temperature and is specified as a function of time as obtained from the TRACG calculations. GEH sensitivity studies showed that {{

}} is conservative for calculating the peak containment pressure.

Additional specifics on modelling parameters and assumptions can be found in the LTR.

Regulatory Commentary
The NRC and CNSC staff determined that the GOTHIC modeling assumptions as discussed in the LTR are conservative; therefore, the NRC and CNSC staff have determined the approach to be acceptable for modeling the BWRX-300 containment temperature and peak pressure.

Nodalization

The LTR presents the details of studies completed by GEH on the effect of nodalization on the GOTHIC results for the containment and the PCCS. The containment study consisted of a base case, coarser grid with double the node size, and two finer grid cases with half the node size in the horizontal and vertical planes as compared to the base case. Figures 6-12 and 6-13 show little difference in the containment pressure, air/steam mixture, and shell temperature results {{ }}, which suggests that the base case nodalization is adequate to resolve the phenomena controlling the containment thermal hydraulic response.

GEH conducted a sensitivity study on a single PCCS unit to understand the sensitivity of the overall PCCS heat transfer to the PCCS nodalization. {{

}}.

Regulatory Commentary

The NRC and CNSC staff determined that the base case containment and PCCS nodalizations are acceptable for use in the evaluation of the BWRX-300 containment response for small and large break LOCAs. Since the modeling uncertainties are adequately addressed through conservative biases and input parameters, further conservatism via the choice of nodalization is unnecessary. The NRC staff also performed confirmatory analyses to verify GEH's conclusions.

Model Uncertainties and Biases

GEH identifies and examines the source of uncertainties in the values of the parameters characterizing the phenomena important to the containment pressure and temperature response. The LTR outlines the conservative biases needed to address the containment modeling uncertainties based on the PIRT results. A summary of the observations made in the LTR are as follows:

1. {{
2.
3.
4. }}
5. the natural and forced circulation and stratification are affected by the friction factors, turbulence modeling, and the model nodalization for the containment and PCCS
6. bounding uncertainties in the convection and condensation heat transfer coefficients need to be accounted for
7. the impact of bounding radiolytic hydrogen and oxygen generation in the RPV and release to the containment needs to be evaluated on the containment response
8. multi-component gas mixture properties need to be accounted for

Regulatory Commentary

The NRC staff independently determined that these observations are appropriate with the exception of number seven (7). Licensing Consideration #1 was developed to address modeling of isolation condenser performance degradation and hydrogen deflagration.

Benchmarking
The LTR provides a high-level overview of benchmarking the GOTHIC code against the Carolinas Virginia Tube Reactor. GEH selected the most applicable test case to the BWRX-300 containment based on the design features, geometry, material, and free volume. The LTR figures demonstrate that GOTHIC, with the appropriate application options and biases, provides a bounding prediction of the Carolinas Virginia Tube Reactor test data.

Regulatory Commentary
The NRC staff independently determined that the conservative options in conjunction with the additional heat transfer biases appropriately bound the Carolina Virginia Tube Reactor test data for pressure and temperature stratification trends. Therefore, this modeling approach is acceptable for use in the BWRX-300 containment evaluation method.

4.5.3 Demonstration of the Method

The LTR presents base cases for the containment response using the GOTHIC model without conservative biases. This provides a reference point for the level of conservatism incorporated into the methodology for the licensing calculations. The LTR provides information to demonstrate the conservatism in the evaluation method for containment response to large and small pipe breaks.

Large Break
The LTR compares the containment pressure and temperature responses for a large steam pipe break LOCA case, calculated using the conservative case assumptions and the base case assumptions. The calculated peak containment pressure shows a reasonable conservatism relative to the base case pressure. GOTHIC predicts that the containment pressure peaks and then decreases over time for both the base and conservative cases.

The LTR also shows the calculated maximum shell temperature to be conservative for the large break case. Additionally, trends in these results correlate to those in the break flow and enthalpy from TRACG, as well as the biases included in the conservative case. They are also consistent with the trends demonstrated by the NRC staff's independent confirmatory analyses.

Additionally, GEH shows a comparison of the containment pressures, heat transfer rates, and containment temperatures results predicted by the biased condensation correlations.

GEH performed extensive sensitivity studies on the break location and determined that {{

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Regulatory Commentary
The NRC and CNSC staff determined that the biases used in the method add a significant conservatism to the peak pressure values, which is sufficient to address the uncertainties in the method.

The described treatment of the limiting break locations for calculating the peak containment pressure and maximum shell temperature is appropriate because it maximizes the conservatisms for the two figures of merit under design basis accident conditions.

Small Break
GEH provided conservative case GOTHIC analysis results for a small steam pipe break and a small liquid pipe break. The LTR showed that the results for the steam and liquid small break LOCA are similar. {{

}}. To ensure this is the most limiting small break LOCA, an application referencing the LTR will analyze both steam and liquid small break LOCA scenarios for peak containment pressure and maximum shell temperature at the respective limiting location and break flow orientation for the BWRX-300.

GEH also performed a break location and flow orientation sensitivity case for a steam small break LOCA at the limiting peak containment pressure and maximum shell temperature break locations identified for the large break LOCA. The results show that {{

}}.

Regulatory Commentary
The NRC and CNSC staff performed independent confirmatory calculations and determined that GEH's sensitivity studies and treatment of small breaks is conservative for estimating the peak pressure and maximum shell temperatures.

5 Joint Review Conclusions

NRC and CNSC staff have determined that the method described in the LTR [1] for using TRACG and GOTHIC to evaluate containment response to LOCAs acceptable provided that the following considerations are addressed prior to licensing:

Licensing Consideration #1	The total volumetric fraction of radiolytic gases in the isolation condenser lower drum should be limited to a sufficiently low level following the event, such that heat transfer is not adversely affected and the hydrogen deflagration margin is maintained for a sufficient period of time.
Licensing Consideration #2	Reverse flow from the RPV into the isolation condenser return line should be prevented for a sufficient period of time following an event, or the entity referencing this method should demonstrate that the TRACG code is capable of conservatively modeling the overall isolation condenser heat removal capacity when reverse flow occurs in the discharge lines.
Licensing Consideration #3	Designs using this method should include a passive containment cooling system that is sized sufficiently large to prevent reverse flow from the containment back to the RPV for a sufficient period of time following the event. The entity referencing this method should demonstrate that reverse flow does not occur, or that any reverse flow that occurs is not safety significant with respect to the acceptance criteria for this method.
Licensing Consideration #4	This method has been demonstrated for a BWRX-300 design with the {{ }} and placement described in the LTR. For any alternate design configuration and placement, the applicability of this method and the PCCS modeling approach should be reviewed for that design.

CNSC staff are currently reviewing the GEH BWRX-300 as part of its VDR process, and the NRC anticipates continuing its preapplication activities on several additional new GEH LTRs. Additionally, GEH is working with both OPG and TVA with regard to the submission of a licence application to construct the BWRX-300 design. As such, both NRC and CNSC staff will continue to review aspects of this design that may relate to the implementation of synergies and efficient design considerations in the future. NRC and CNSC could have additional assessment considerations that arise for needed discussion. One such follow-up item that has been identified by CNSC as a result of its review is as follows:

- TRACG predicts flow oscillations in the ICS pool after the peak containment pressure is reached during a large break LOCA. These flow oscillations do not have an impact on the peak containment pressure analyses. However, the oscillations should be understood and explained for other transient analyses.

It should be noted, upon implementation of this methodology into a specific licensing application of the BWRX-300 design, the NRC and CNSC staff will evaluate the items identified in this joint report and their interfaces with the proposed licence application to ensure consistency. Each organization will make its own regulatory determinations regarding the topics identified in this report, as applicable.

6 References

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