**SAFETY EVALUATION BY THE OFFICE OF NUCLEAR REACTOR REGULATION**

**LICENSING TOPICAL REPORT NEDC-33922P, REVISION 2**

**BWRX-300 CONTAINMENT EVALUATION METHOD**

**GE-HITACHI NUCLEAR ENERGY AMERICAS, LLC**

**DOCKET NO. 99900003**

## 1.0 INTRODUCTION

The purpose of GE-Hitachi Nuclear Energy Americas, LLC (GEH), Licensing Topical Report (LTR) NEDC-33922P, Revision 0, "BWRX-300 Containment Evaluation Method," dated September 25, 2020 (Agencywide Documents Access and Management System (ADAMS) Accession No. ML20269A469), supplemented November 19, 2021 (ADAMS Accession No. ML21323A008) and updated in Revision 2, dated December 17, 2021 (ADAMS Accession No. ML21351A168), is to present an acceptable analysis method for BWRX-300 containment thermal-hydraulic performance to demonstrate that the containment design satisfies the acceptance criteria documented in the approved LTR NEDC-33911P-A, Revision 3, "BWRX-300 Containment Performance," dated January 7, 2022 (ADAMS Accession No. ML22007A021). Specifically, Section 3.0 of NEDC-33911P-A, Revision 3, outlines the scope of the acceptance criteria for the containment analysis method for the BWRX-300 design-basis events (DBEs) to show that the containment performance analysis meets the acceptance criteria for the following:

- anticipated operational occurrences (AOOs)
- station blackout as required by Title 10 of the *Code of Federal Regulations* (10 CFR) 50.63, "Loss of all alternating current power"
- anticipated transients without scram (ATWS) as required by 10 CFR 50.62, "Requirements for reduction of risk from anticipated transients without scram (ATWS) events for light-water-cooled nuclear power plants"
- large-break loss-of-coolant accident (LBLOCA) inside containment
- small-break loss-of-coolant accident (SBLOCA) inside containment

In this safety evaluation (SE), the U.S. Nuclear Regulatory Commission (NRC) staff describes its review and the acceptability of the BWRX-300 containment evaluation method (CEM) proposed for the BWRX-300 small modular reactor (SMR) using the Transient Reactor Analysis Code General Electric (TRACG) code for the reactor pressure vessel (RPV) model and the Generation of Thermal-Hydraulic Information for Containments (GOTHIC) code used separately for the containment model as described in LTR NEDC-33911-A, Revision 2, Section 3.3 and Section 3.4. This review included NRC staff requests for additional information (RAIs) that GEH responded to in letters dated May 19, 2021 (ADAMS Accession No. ML21139A110); September 17, 2021 (ADAMS Accession No. ML21260A010); October 8, 2021 (ADAMS Accession No. ML21281A081); October 29, 2021 (ADAMS Accession No. ML21302A080); and December 17, 2021 (ADAMS Accession No. ML21351A168). Additionally, the staff conducted a regulatory audit from January 5, 2021 to December 8, 2021 (ADAMS Accession No. ML20363A025). The staff detailed audit activities and review can be found in the audit report (ADAMS Accession No. ML21343A194).

The NRC staff will evaluate the compliance of the final BWRX-300 SMR design including the containment and the final CEM during future licensing activities in accordance with 10 CFR Part 50, "Domestic Licensing of Production and Utilization Facilities," or 10 CFR Part 52, "Licenses, Certifications, and Approvals for Nuclear Power Plants," as applicable.

The staff evaluated the applicable regulations and guidance for the BWRX-300 CEM in Section 3.1 of the staff SE as part of the approved LTR NEDC-33911P-A, Revision 3. The staff views this SE as a continuation of LTR NEDC-33911P-A, Revision 3 and, as such, the same regulatory evaluation applies herein. However, for ease of reference, regulations related to CEM are as follows:

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

- 10 CFR Part 50, Appendix K, "ECCS Evaluation Models," as it relates to sources of energy during the LOCA, provides requirements to assure that all the energy sources have been considered.

- 10 CFR 50.44, "Conditions of License," as it relates to boiling-water reactors (BWRs) and pressurized-water reactors (PWRs) being designed to accommodate hydrogen generation equivalent to a 100-percent fuel clad-coolant reaction; to limit containment hydrogen concentration to no greater than 10 percent; to have a capability for ensuring a mixed atmosphere during design-basis and significant beyond-design-basis (BDB) accidents (a significant beyond-design-basis accident (BDBA) being an accident comparable to a degraded core accident at an operating (as of October 16, 2003) light-water reactor in which a metal-water reaction occurs involving 100 percent of the fuel cladding surrounding the active fuel region (excluding the cladding surrounding the plenum volume)); and to provide containment-wide hydrogen control (such as igniters or inerting), if necessary, for certain severe accidents, noting that post-accident conditions should be such that an uncontrolled hydrogen/oxygen recombination would not take place in the containment, or the plant should withstand the consequences of uncontrolled hydrogen/oxygen recombination without loss of safety function or containment structural integrity.

- 10 CFR 50.46, "Acceptance criteria for emergency core cooling systems for light-water nuclear power reactors," as it relates to requirements for long-term cooling, including adequate net positive suction head margin in the presence of LOCA-generated and latent debris

## 2.0 OVERVIEW OF THE BWRX-300 REACTOR PRESSURE VESSEL AND CONTAINMENT FEATURES PERTINENT TO THE APPLICATION METHOD

LTR Section 2.0 provides high-level information about the BWRX-300 CEM. Previously approved LTRs NEDC-33910P-A, Revision 2, “BWRX-300 Reactor Pressure Vessel Isolation and Overpressure Protection,” June 2021 (ADAMS Accession No. ML21183A259) and NEDC-33911P-A, Revision 3, describe the BWRX-300 RPV isolation and overpressure protection and BWRX-300 containment performance, respectively.

The following containment design features are relevant to the purposes of the LTR:

- The containment is a dry enclosure, near atmospheric pressure during normal operation.
- The containment design pressure and temperature are within the experience base of conventional boiling-water reactors (BWRs).
- The containment is inerted with nitrogen during normal operation.
- There are no subcompartments containing large bore high energy lines.
- The subcompartments have sufficiently large openings such that the boundaries of the subcompartments do not experience large pressure differentials resulting from pipe breaks outside the subcompartments.
- The passive containment cooling system (PCCS) is a [[

]].

## 3.0 LOSS-OF-COOLANT ACCIDENT SCENARIOS AND LIMITING PIPE BREAKS

LTR Section 3.0, describes DBEs of pipe breaks inside containment and break scenarios to establish the limiting pipe breaks for the containment performance evaluation. The postulated breaks include large breaks of main steam pipes, feedwater pipes, isolation condenser system (ICS) steam supply and condensate return pipes, reactor water cleanup (RWCU) pipes, and shutdown cooling (SDC) pipes, and all unisolated small steam and liquid pipe breaks. All the large breaks listed above have RPV isolation valves, which limit the amount of mass and energy (M&E) release from the RPV to the containment.

For the CEM described in the LTR, the outboard containment isolation valves are assumed to remain open for a main steam pipe break, but the turbine stop valve is closed instantaneously, concurrent with the break. Because the two steam pipes are connected to a common header upstream of the turbine stop and control, the intact loop also contributes to the break flow rate before the RPV isolation valves are closed. Using this set of assumptions maximizes the discharge of steam to containment.

As described in the LTR, the feedwater pipes have a check valve outside containment. As a result, the intact feedwater loop does not feed backward to the break. The feedwater pump trips on a pipe break. The hot water in the feedwater piping may flash and contribute to break flow. The isolation condenser (IC) steam supply pipe flow area is smaller than the main steam pipe flow limiter flow area. The IC condensate return pipe flow area is much smaller than the feedwater pipe; therefore, a break in the condensate return piping in an IC train is bounded by the feedwater pipe breaks. Further, pipe inventory in the main steam piping is much larger than the inventory in an IC train and its attached piping. Considering the pipe size, configuration, and isolation signal timing assumed in the analysis, a break in the ICS piping is bounded by a break in the main steam and/or feedwater pipe as analyzed.

The LTR states that, because the RWCU and SDC pipes are smaller in diameter than the feedwater pipes, and the isolation valve closure timing is the same for all pipes, breaks in these pipes are also bounded by feedwater pipe breaks.

The applicant stated in the LTR that from the spectrum of breaks the limiting large breaks are:

- main steam pipe
- and the feedwater pipe

The limiting small breaks are unisolated instrument line breaks, either in the steam or liquid space. The inside diameters of the instrument lines are [[
]].

LTR NEDC-33911P-A, Revision 3, Section 3.1, states that the containment design will be based on consideration of a full spectrum of DBEs that would result in the release of reactor coolant to pressurize the containment. These containment DBEs include liquid and steam breaks and will be evaluated using the TRACG code as a boundary condition to the GOTHIC code to calculate the containment response.

The NRC staff finds that the description in LTR Section 3.0 is consistent with the scope of the evaluation model as described in LTR NEDC-33911P-A, Revision 3, and addresses the full spectrum of DBEs being included to determine the limiting breaks for the containment analyses. When the BWRX-300 CEM is applied to the final design, the spectrum of breaks will be used in the containment analyses to determine M&E and the resulting containment pressure and temperature.

## 4.0 OVERVIEW OF THE EVALUATION MODEL

The BWRX-300 CEM is based on using TRACG code to evaluate the M&E from the RPV, and the GOTHIC, code to evaluate the resulting containment response. The M&E calculated by the TRACG RPV model is used as a boundary condition in the GOTHIC containment model used to calculate the containment pressure and temperature response. The M&E model for the BWRX-300 containment response uses the applicable parts of NEDC-33083P-A, Revision 1, “TRACG Application for ESBWR,” issued October 2010 (ADAMS Accession No. ML102800567), which is incorporated in the approved Economic Simplified Boiling-Water Reactor (ESBWR) design certification (DC), certified by the NRC in 2014 (10 CFR Part 52, Appendix E, "Design Certification Rule for the Economic Simplified Boiling-Water Reactor.") Section 5.0 of this SE discusses the acceptability of this application of the ESBWR TRACG method to the BWRX-300 for calculation of the M&E to containment.

The LTR presents the standalone GOTHIC containment model developed for the BWRX-300. GOTHIC is a commercially available computer code that has been developed consistent with regulatory requirements in 10 CFR Part 50, Appendix B, and meets the GEH software quality requirements. The containment analyses in this LTR are performed using the latest GOTHIC version 8.3. 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 [the] BWRX-300 containment application caused by any code changes can be successfully dispositioned." The final SE for GEH LTR NEDE-32906P-A, Supplement 3, "Migration to TRACG04/PANAC11 from TRACG02/PANAC10 for TRACG AOO and ATWS Overpressure Transients" dated April 16, 2010 (ADAMS Accession No. ML110970401) shows that the NRC staff had allowed similar usage of updated versions of the TRACG code maintained under the quality assurance criteria of 10 CFR Part 50, Appendix B. GEH also stated that they intend to use any future GOTHIC versions under 10 CFR 50.59 process. The staff evaluated GEH's position regarding the use of future versions of GOTHIC code and finds it to be acceptable and consistent with NRC requirements controlling changes made without requiring prior NRC review and approval.

The LTR states that the "BWRX-300 containment design is much simpler than the ESBWR containment, and many of the ESBWR containment phenomena do not apply to the BWRX-300 containment. Phenomena that are of secondary importance to the ESBWR containment response may become important to the BWRX-300 containment response." The staff has reviewed the information provided in LTR Sections 6.1, 6.2, and 6.4 about the GOTHIC phenomenon identification and ranking table (PIRT), the PIRT survey, and the knowledge level for the phenomena pertinent to containment analysis has been reviewed and the staff has documented its findings in the corresponding sections in this SE.

The LTR states that "the evaluation method for the BWRX-300 containment response to DBEs has been developed following the applicable elements of Regulatory Guide (RG) 1.203, "Transient and Accident Analysis Methods," issued December 2005 (ADAMS Accession No. ML053500170)." In Section 6.3 of this SE, the staff discusses the acceptability of the assessment basis for the BWRX-300 containment model using RG 1.203.

## 5.0 TRANSIENT REACTOR ANALYSIS CODE GENERAL ELECTRIC METHOD FOR MASS AND ENERGY RELEASE

### 5.1 Transient Reactor Analysis Code General Electric Code and Qualification

The TRACG code is the GEH proprietary version of the Transient Reactor Analysis Code (TRAC) used for best-estimate analysis of BWR transients ranging from AOOs to design-basis LOCAs and ATWS events. This code has been previously reviewed and found acceptable for applications of current operating BWRs, the Simplified Boiling-Water Reactor (SBWR), and the ESBWR, based on results of separate effects tests, component performance tests, integral effects tests, and full-scale plant data as described in the following qualification reports; (1) NEDC-32725P, Revision 1," General Electric Company TRACG Qualification for SBWR, Vol. 1 & 2, for Pre-Application Review of ESBWR, issued September 2002 (ADAMS Accession No. ML022560081); (2) NEDC-33080P, Revision 1, "TRACG Qualification for ESBWR," issued May 2005, (ADAMS Accession No. ML051600373); (3) NEDE-33005P-A, Revision 2, "Licensing Topical Report, TRACG Application for Emergency Core Cooling Systems / Loss-of-Coolant-Accident Analyses for BWR/2-6," issued May 2018 (ADAMS Accession No. ML18143A214);

and (4) NEDC-33083P-A, Revision 1. Therefore, the staff focused its review on changes made to the code and modeling methods that postdate the ESBWR DC, as well as the application of this method to the BWRX-300 design. As such, the staff determined that the M&E rate is a crucial component of the containment analysis where the focus is on conservatively biasing the M&E to determine the peak containment pressure (PCP) and temperature.

The applicant performed extensive qualification and application studies (NEDC-33080P and NEDC-33083P-A, Revision 1) for the ESBWR design due to higher power and other design differences as compared to the SBWR. The applicant, however, did not perform similar in-depth studies of the BWRX-300 since the applicant considers it to be a scaled down version of the ESBWR design. Upon review, the staff largely agrees with this conclusion (specifically with respect to the reactor vessel and its internal systems/components), however, the designs have several key differences in safety system components. The containment "dry" design and other redesigned supporting safety systems described in NEDC-33910P-A, Revision 2, for the BWRX-300 result in a significantly different transient progression and response compared to those evaluated for the ESBWR design. Therefore, the staff focused its review on these aspects where the PIRT consideration of application of the TRACG code to the ESBWR may not completely address the phenomenological progression of the LOCA event for the BWRX-300.

To determine whether using the M&E evaluation model described in the LTR results in an acceptable prediction of the M&E release, the NRC staff reviewed the various phenomena modeled in the LTR and considered whether they were modeled appropriately. These key phenomena are different for the large and small break assessments. For the large break, [[ ]] before the RPV isolation valves close, which prevents further M&E release to containment. Reactor cooldown is then managed by the ICS and [[ ]]. In this method, M&E from each side of the break, before RPV isolation, is summed as a source input for the GOTHIC analysis. In addition, M&E from the intact loop continues for a short time to drain steam and condensate remaining in the steam lines.

For small breaks [[ ]], there are no isolation valves, so the M&E releases [[ ]] throughout the 72-hour transient used for passive plant designs. Consequently, the RCS inventory loss during small breaks can result in reactor core water levels dropping below the top of active fuel (TAF), creating the potential for fuel cladding heat up.

The applicant conservatively assumed in the TRACG code analysis that back pressure from the containment will remain at atmospheric pressure to maximize the M&E release. The staff agrees that this decoupled method will result in higher break flow rates than if the analysis considered pressure increase in the containment. However, since the break flow remains mostly choked, the inclusion of containment back pressure would have minimal to no effect on total M&E from the RCS during the blowdown period.

The staff understands that for this design the LOCA blowdown is very different from that for typical large light-water reactors (LWRs). This is due to the much smaller pipe sizes and the number and extent of passive safety systems employed. As mentioned, GEH did not perform a specific PIRT for the BWRX-300 TRACG code M&E method in accordance with RG 1.203 but relied on previous ESBWR evaluations to determine the important phenomena and the appropriate uncertainties to be applied. The applicant details these differences with the

BWRX-300, which affect modeling inputs, in Section 5.2 of the LTR and concludes that they remain within the same qualification range of conditions as the ESBWR.

RG 1.203 indicates that fidelity in an evaluation model is related to the existence and completeness of validation efforts. Although the staff agrees that the preliminary BWRX-300 design described in the LTR shares many similarities with the ESBWR design, the applicant did not present a complete review of differences related to the new safety system functions in the LTR. However, the staff notes that such a review is not necessarily required for the limited scope of this LTR to determine the maximum M&E to containment from a postulated LOCA, which involves primarily a RPV blowdown, calculation of critical flow, and the longer-term RCS cooldown. Based on the limited scope of this LTR and expected similar behavior of the M&E release modeled in this LTR to the ESBWR qualification range, the staff evaluated the similarities/differences and determined that use of the TRACG code and the qualification basis performed for the ESBWR as shown in Section 3.3 of LTR NEDC-33911P-A, Revision 3, is acceptable for this use. The staff determination however, is limited to the extension of the PIRT application for the purpose of M&E for containment analysis only.

## 5.2 Application of the ESBWR TRACG Loss-of-Coolant Accident Method to BWRX-300 Mass and Energy Release Calculations

### 5.2.1 Transient Reactor Analysis Code General Electric Reactor Pressure Vessel Nodalization for BWRX-300

The GEH TRACG M&E RPV model for the BWRX-300 described in the LTR is a modification of the previously approved M&E model described in NEDC-33080P, Revision 1, developed for the much larger ESBWR. Since the vessels are similar in height, the axial nodalization is very similar to ESBWR. The radial nodalization is increased significantly in the core region, even though the core diameter is smaller, while one region is still used for the downcomer. The applicant indicated that this refinement allows the use of the same RPV model both in safety analysis transients and LOCA analyses; however, the staff only reviewed this model for its use for M&E analysis.

Related piping for main steam and feedwater systems are attached to the RPV and scaled from the ESBWR with RPV isolation valves added. The model includes the ICS noding, but the BWRX-300 ICS performs a different function than in the ESBWR design, since the ICS alone provides overpressure protection and emergency core cooling for the BWRX-300 design (as described in NEDC-33910P-A, Revision 2). The ICS has three loops, but only two of the three are modeled as one is assumed unavailable for the licensing-basis single failure criterion. The redesigned BWRX-300 PCCS is not modeled in TRACG and no other containment systems are modeled. For the ESBWR design, both RCS and containment wet well, suppression pool, and drywell systems were modeled within TRACG, but since the BWRX-300 design uses a dry containment, the applicant elected to use a standalone GOTHIC calculation to determine containment peak pressurization from the LOCA transients.

The model contains break noding for main steam and feedwater and small liquid and steam instrument line breaks as shown in LTR Figure 5-3. The ICS supply and return lines, steam box, tubing, water box, and return valves are modeled as shown in LTR Figure 5-4. Each ICS is contained [[ ]] of the reactor cooling pool (LTR Figure 2-2). The ICS noding is more detailed than that used for the ESBWR design. The applicant indicated this is, in part, the reason for code convergence issues related to deficiencies in modeling transport of radiolytic gases.

Main feedwater pumps, isolation valves and specific piping segments are used to model inlet piping at the elevation near the top of the chimney as shown in LTR Figure 5-2. [[

]]. The steam lines include the RPV isolation valves and piping modeled back to the turbine control valve.

Based on GEH's refinement of the modeling of the BWRX-300 in relation to the previously approved ESBWR as described in LTR Section 5.2.1 and appropriate modeling of the key design features of the plant, the staff determined that the TRACG RPV nodalization is acceptable for determining the M&E release.

### 5.2.2 Large and Small Pipe Breaks

The LTR states that the large steam line break modeling uses piping segments and break location nodes to simulate a double-ended guillotine break in piping just downstream of the RPV isolation valves, as shown in LTR Figure 5-3. [[

]].

For each of these large breaks, there are RPV isolation valves mounted directly to the vessel that [[

]]. In future licensing applications, the applicant would be expected to conservatively estimate the timing input for the drywell pressure signal for each break location.

The ICS valves for a minimum number of IC trains are also assumed to begin opening [[
]] after the transient. One train is assumed for steam line break and two are assumed available for the feedwater line break cases. The applicant chose this additional conservatism to ensure that the main steam pipe break analysis is bounding as compared to an ICS steam pipe break, which would have fewer IC trains available for decay heat removal.

Reactor trip is assumed due to [[

]]. During the regulatory audit discussions, the staff sought clarification from the applicant as to whether reactor power will increase just after break initiation if there were a concurrent turbine trip. GEH indicated that the power was modeled conservatively to account for the potential increase in power by the [[
]] is built into the decay heat curves. The staff agreed that

[[ ]] would reasonably bound the power surge resulting from a turbine trip pressure wave.

The small steam line break is modeled as a valve directly off the RPV to simulate the break opening, in a single sided configuration, discharging into a break node that is used to represent the containment. These are modeling unisolable RPV instrument taps, assumed [[ ]] in diameter, with the upper tap used for the steam break and the lower one for the liquid break. The small break transients each use two IC trains, with one unavailable due to single failure criteria. Consistent with the large-break modeling, the containment is assumed to remain at atmospheric pressure to maximize the break flow rate.

NUREG-0800, "Standard Review Plan for the Review of Safety Analysis Reports for Nuclear Power Plants: LWR Edition," the Standard Review Plan (SRP), Section 6.2.1.3, "Mass and Energy Release Analysis for Postulated Loss-of-Coolant Accidents (LOCAs)," Revision 3, issued March 2007 (ADAMS Accession No. ML053560191), provides guidance for the review of M&E release analysis from postulated LOCAs. The thermal-hydraulic modeling and nodalization used for the BWRX-300 must be adequate to predict all important phenomena in accordance with RG 1.203. Based on this limited scope of the LTR and the expected similar behavior of the M&E release modeled in the BWRX-300 to the ESBWR qualification range, the staff determined that the TRACG code modeling and assumptions used as shown in LTR Section 5.2.1 are acceptable.

### 5.2.3 Channel Grouping, Decay Heat and Power Shape

GEH states that the BWRX-300 core is modeled using a [[ ]]The outer [[ ]]. The core is modeled at 887.4 megawatts thermal (MWth) using a generic Global Nuclear Fuel (GNF) bundle type and standard-length core, similar to the ESBWR for the conservative cases. The core is modeled [[ ]]. The operating and decay heat [[ ]] by the applicant, consistent with NEDE-33005P-A, Revision 2. The staff found that the BWRX-300 core modeling was consistent with previously approved methods and therefore acceptable.

As identified in LTR Table 5-4, the applicant uses American Nuclear Society (ANS) 5.1-1979, "Decay Heat Power in Light Water Reactors," nominal as the decay heat curve to compute decay heat power for the base cases, and ANS 5.1-1979 plus 2 sigma for the conservative cases. SRP Section 6.2.1.3 provides guidance for the review of the M&E release analysis from postulated LOCAs and endorses ANS-5 1971, "Decay Energy Release Rates Following Shutdown of Uranium-Fueled Thermal Reactors," decay heat curve, to meet the requirements of 10 CFR Part 50, Appendix K, paragraph (I)(A) "Sources of heat during the LOCA," to calculate energy available for M&E releases. However, in response to RAI 06.02.01.03-01, dated May 19, 2021, GEH provided a description of the BWRX-300 M&E method's compliance with Appendix K (I)(A) and referenced consistency with ESBWR (NEDC-33083P-A, Revision 1) as justification for ANS 5-1 1979 decay heat method.

Based on staff's previous analysis in the ESBWR DC approval and recent review of Westinghouse containment method (WCAP-17721P-A, "Westinghouse Containment Analysis

Methodology," August 2015 (ADAMS Accession No. ML15221A008)), the staff agrees that the use of ANS 5.1 1979 with 2 sigma is appropriately conservative and acceptable for use in the TRACG code M&E method for the BWRX-300 design.

Additionally, the staff reviewed the other requirements of Appendix K, paragraph (I)(A) including (1) initial core power and RPV stored energy, (2) fission heat, (3) decay of actinides, (4) metal-water reaction rate, including the rate of energy release, hydrogen generation, and cladding oxidation from the metal/water reaction, and (5) reactor internals heat transfer. The staff found that the BWRX-300 design's modeling was consistent with these Appendix K requirements. It should be noted that although the TRACG code has the capability to compute energy from metal/water reactions, GEH has not considered the potential generation of hydrogen due to metal- water reactions. The staff agrees this is acceptable since the BWRX-300 LOCA acceptance criteria limits cladding temperatures (LTR Section 5.2.3) to those of normal operation such that the temperature threshold of the metal/water reaction would not be reached.

### 5.2.4 Isolation Condenser Modeling and Radiolytic Gases

The ICs are a similar design to those used for ESBWR; however, the function and connections to the reactor are quite different, as described in NEDC-33910P-A, Revision 2. LTR Figure 5-4 shows the modeling of the ICs. The system consists of [[

]].

In accordance with LTR Figure 2-3, [[

]].

The staff noted that In typical BWR reactor systems such as the BWRX-300, radiolytic gases (hydrogen ($H_2$) and oxygen ($O_2$)) 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 IC tubing as the ICs begin operation. Then during the condensation process, these noncondensable gases (NCGs) would begin to accumulate in the lower drums and tubing. The accumulation and buildup of NCGs begins at the steam-water interface in the lower drum, and then continues to expand into the tube region, where the bulk of condensation is occurring. In its review, the staff considered that this buildup of NCG in the tubing can significantly degrade the ICs ability to condense steam, reducing its heat transfer capacity to cool the reactor. This buildup of NCG as related to IC performance degradation is critically important for SBLOCA cases where the break is unisolated [[

]]. The IC heat transfer performance must be adequate to cool and depressurize the reactor and to maintain a break flow low enough so that sufficient inventory remains within the reactor core region.

As such, the staff focused on the sufficiency of the method in accounting for the impact of the presence of NCGs. The IC 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. Its safety function is also expanded to include [[ ]] long-term cooling within a new dry containment system. [[

]]. For this LTR analysis, the large break cases assume the high drywell pressure occurs at [[ ]] and the small break cases assume high drywell pressure at [[ ]]. The transport, intrusion, and buildup of NCG into the ICs is not explicitly modeled by GEH in the BWRX-300 model. During the staff's regulatory audit discussions, GEH stated [[

]].

The formulation used for generation of radiolytic gases per MWth was given in LTR Section 5.2.4 as a bounding value used generically for the BWR fleet. This volumetric rate was reviewed and approved by NRC staff in a previous submittal (NEDC-33004P-A, Revision 3, "Constant Pressure Power Uprate," dated March 2003 ADAMS Accession No. ML031190318), and thus judged suitable for application to the BWRX-300. The staff notes that this rate is based on post-trip off gas measurements and does not include or address hydrogen that could be generated from potential cladding oxidation, which is not expected to occur in the BWRX-300 based on the M&E LOCA criteria set forth in LTR NEDC-33911P-A, Revision 3.

The applicant then used a series of control block calculations in the BWRX-300 model to estimate radiolytic gas progression into the ICs for determining an overall percentage accumulation in the ICs. The standalone model included simulation of mechanistic transport of $H_2$ and $O_2$ within the code but with idealized boundary conditions used from the RPV steam supply and return line connections. [[

]]. The standalone model was run at a range of pressures and NCGs concentrations to develop degradation curve fits. These were then used in the BWRX-300 model to simulate IC tube heat transfer degradation via reduction of a PIRT multiplier based on the estimated accumulation of NCGs in the control blocks.

Because of the idealized boundary conditions used in the standalone model, the control block method will underpredict the accumulation of NCGs in the ICs and therefore the amount of IC heat transfer degradation. The applicant provided additional information in its response to RAI 06.02.01.03-01, dated May 19, 2021, on the conservatisms of the PIRT model used and further discussed degradation effects of the gradual accumulation of NCGs in ICs tube bundles, the associated uncertainties and the subsequent consequences for both the large break and small break limiting cases. The applicant acknowledged that [[

]].

The staff also reviewed the GEH response to RAI 06.02.01-01, dated May 19, 2021, regarding sensitivity analysis results performed by the applicant that demonstrated that [[

]].

The staff also considered the fact that the BWRX-300 design is not finalized, so the ultimate demonstration that [[
]]. Therefore, to enable the staff to reach the finding that the model

produces acceptably conservative results, the staff-imposed a Limitation and Condition (L&C) number (#) 1. documented in Section 7.0 of this SE, such that the total volumetric fraction of radiolytic gases in the IC lower drum be controlled to a sufficiently low level.

The staff noted that the IC return pipe layout for the BWRX-300 design is different from that used for the ESBWR design, and that [[

]].

During a SBLOCA, the chimney steam volume is at a higher pressure than the ICs and steam dome regions, so that at the discharge point, the ICs discharge pipe could face reverse steam flow toward the lower drum when the condensate flow rate is low and the system pressure is reduced. The staff identified that [[

]]. The applicant confirmed in its response to RAI 06.02.01-06 dated October 8, 2021, that the original ESBWR design's return line loop seal would be retained in the final design of the BWRX-300. The applicant acknowledged that large direction dependent form losses were added to the discharge into the chimney to represent the effect of a water loop seal. During the regulatory audit the staff confirmed that the periodic flushing through of steam that would still occur should not significantly degrade the IC performance since accumulation of NCGs are being controlled to a low concentration in the IC lower drums through L&C # 2. documented in Section 7.0 of this SE. [[

]].

The applicant confirmed in its RAI response to Question 06.02.01-01 dated May 17, 2021, that [[

]]. The staff also finds that the applicant's conclusion [[

]] is reasonable, and that it is not necessary to address these phenomena in the containment method. Therefore, the staff found that the method is adequate [[ ]].

### 5.2.5 Modeling Biases Phenomenon Identification and Ranking Tables

The TRACG code has built-in modeling bias parameters, called PIRTs (e.g., critical flow), that can be specified in code input to modify the results of code runs. TRACG PIRTs are placed strategically on important coefficients and correlations to add or remove biases. This is done with the objective of adding conservatisms or uncertainty locally, or globally, to internally calculated nominal code values. The PIRT ranges 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 (NEDC-33083P-A, Revision 1).

These PIRT multipliers are related to but not the same as the PIRT process. In general, the PIRT process, per RG 1.203, is one of the key steps used to develop an evaluation methodology (EM) or a calculational framework for evaluating the behavior of a particular reactor system for postulated transient or design-basis accident (DBA) analysis. The applicant did not develop an EM specifically for the BWRX-300 design, but rather is depending on and refers to previous evaluations performed for the ESBWR as being directly applicable to this design.

The BWRX-300 is comparable in height to the ESBWR design, with the vessel radius, core flow area, quantity of fuel, and other internals scaled down from the ESBWRs 1,520 MWe to 300 MWe. With similar vertical internal features and apportionments, the staff agrees that internally the BWRX-300 is very similar to the ESBWR. However, outside of the reactor vessel, the safety systems and equipment, including the containment, are very different than the ESBWR design. Besides the new dry containment, the major safety equipment differences are related to using [[ ]] (NEDC-33910P-A, Revision 2). Therefore, the staff notes that the PIRT and modeling biases developed for the ESBWR are not all applicable to the BWRX-300 as is indicated, in part, by LTR Table 5-1. As such, the staff performed a sensitivity analysis of the relevant PIRT modeling parameters listed in LTR Table 5-1 and determined that there was negligible impact on the results and that they were conservatively applied for this M&E analysis application for the BWRX-300.

The PIRT multiplier used in the BWRX-300 TRACG LOCA M&E analysis [[ ]]. The staff also evaluated the critical flow modeling for consistency with SRP Section 6.2.1.3 and 10 CFR Part 50, Appendix K, discussed in SE Section 5.3.2. In addition, the staff reviewed the conservatism in decay heat modeling that is based on ANS 5.1-1979 with 2 sigma uncertainty, consistent with ESBWR usage, and that evaluation was presented earlier in SE Section 5.2.3.

### 5.2.6 Initial Conditions for Base and Conservative Cases, Trips

The initial conditions used in the base and conservative cases are listed in LTR Table 5-2. These key initial conditions are considered important plant operating parameters describing steady-state nuclear and hydraulic conditions from which a transient is initiated. The reactor power is at 102 percent of the nominal 870 MWth and the steam and feedwater flow are increased commensurately. The steam dome pressure is increased by [[ ]] and the hot channel power profile is top peaked. The downcomer water level is modeled as [[ ]]. Since the RPV isolation valves close rapidly upon detection of a large break, the initial water level has very low significance for the large break cases. As such, the staff found that modeling of the initial RPV conditions and the valve closure timing inputs are acceptable.

Since this is a methodology LTR, the exact reactor trip signals are not modeled in the example cases provided, but reasonable times and events are assumed at transient initiation like loss of alternating current (AC) power at break initiation. The reactor scram delay is conservatively set at [[ ]] is built into the scram table. The high containment pressure trip, which actuates the ICs, is manually set at [[ ]] for the large break cases, and [[ ]] for the small break cases. Main feedwater coasts down to [[ ]] and postulated break opens instantly [[ ]]. High containment pressure or water level at Level 2 causes RPV isolation valves to close, isolating all connections [[ ]] or larger. For the LTR, it is assumed [[ ]]. The IC valves are actuated by [[ ]].

Due to single failure, generally only two of three ICs are assumed to be available for the M&E LOCA analyses. However, for the large steam break only one of three ICs is assumed available. The applicant added this conservatism to the large break steam case so that it remained limiting in comparison to an assumed IC supply line break case which would also assume only one IC is available. Also, the containment backpressure for the TRACG M&E releases is conservatively assumed to remain at atmospheric pressure throughout the transient.

The staff reviewed these initial conditions for the base and conservative cases, particularly LTR Table 5-1, and determined they were correctly selected and biased, as necessary, to produce appropriately conservative results. The staff's own independent sensitivity studies that used the TRAC/RELAP Advanced Computational Engine (TRACE) best-estimate reactor systems code developed by the NRC for analyzing transient and steady-state thermal-hydraulic behavior in LWRs supports this finding. Therefore, the staff agrees that the initial conditions outlined in the LTR are acceptable for use.

## 5.3 Demonstration Cases for Large Breaks

### 5.3.1 Base Case for Large Feedwater and Steam Breaks

The main steam pipe breaks are analyzed conservatively to bound all other steam pipe breaks by assuming only one IC is available. The base cases use nominal initial conditions and nominal modeling parameters, (i.e., all the PIRT multipliers described in LTR Table 5-1 are set to 1.0). The staff reviewed the results of the base cases because they provide an indication of margin added to key figures of merit (FOM) by the conservative analysis.

The large steam break progresses as follows:

- Main steam pipe break occurs inside the containment concurrent with loss of offsite power.

- Feedwater pumps trip [[ ]].

- A reactor trip occurs [[ ]].

- Drywell high pressure is reached at [[ ]].

- [[ ]].

- The IC return valves for IC train A open at [[ ]].

The transient is run out to [[ ]], which allows the RPV to cooldown to near atmospheric pressure. The staff noted that [[ ]].

Main feedwater pipe breaks are modeled [[ ]]; otherwise, the assumptions used are the same as those for the large steam break. The progression of the feedwater line break have the following differences:

- The main feedwater pipe break occurs inside the containment.

- [[ ]].

The staff determined that the base cases are modeled appropriately, and that the overall conservatism of the conservative cases is sufficient. The staff reached this conclusion by reviewing the base cases, including information obtained during the regulatory audit, and confirming this conservatism with the staff's independent calculations using TRACE.

### 5.3.2 Conservative Case for Large Feedwater and Steam Breaks

The staff reviewed the LTR basis documents and calculations during the regulatory audit for the conservative cases of the large feedwater and steam breaks. These cases use initial power conditions with [[ ]] so main steam and feedwater flow are increased, and conservative modeling parameters, (i.e., the PIRT multipliers described in LTR Table 5-1, [[ ]]. The initial conditions are based on LTR Tables 5-2 and 5-4 inputs for conservative calculation cases. The applicant indicates in Section 5.2 of the LTR that the modeling inputs used for the BWRX-300 are within the qualification and assessment ranges performed for the ESBWR design, so that the PIRTs are directly applicable as previously indicated and reviewed in Section 5.2.5. [[ ]]. Consequently, the conservative cases reach a worse outcome in terms of the FOM, so these are the cases the staff evaluated to determine acceptability of the method consistent with SRP Section 6.2.1.3 and 10 CFR Part 50, Appendix K.

The break mass flow for the conservative main steam line break case is shown in LTR Figure 5-13 and this case uses a [[ ]] on critical flow. [[ ]]. The staff reviewed the mass flow generated by this input and found it to be reasonably conservative in comparison to results obtained by the staff calculations, using the Moody correlation (ADAMS Accession No. ML12142A162) extrapolations, and confirmatory TRACE code results. The staff therefore determined that the mass flow results are acceptable and consistent with meeting the requirements of 10 CFR Part 50, Appendix K, and Appendix A, GDC 50.

The staff also checked a sampling of the other PIRT settings to ensure they yielded conservative results in comparison to settings used in Table 5-1 of the LTR. The staff found that besides the PIRT multiplier for critical flow, [[ ]]. The staff's analysis with TRACG code confirmed that the PIRT values have a very negligible effect on main steam break cases.

The staff also reviewed the main feedwater event results shown in LTR Figures 5-15 through 5-17 and determined that, the mass entering containment is slightly larger than the steam line break, but the energy released is significantly less, so it is not limiting for M&E releases. [[ ]]. Both the large main feedwater and main steam transients retain adequate core inventories so that the core is never uncovered. The staff found that the modeling and results of the main steam and feedwater cases are reasonable and acceptable for determining M&E releases for the BWRX-300 design considering L&C # 1. as documented in Section 7.0 of this SE.

### 5.4 Demonstration Cases for Small Breaks

The staff reviewed the LTR and the basis documents and calculations during the staff regulatory audit regarding the small steam and liquid pipe breaks. The small steam and liquid pipe breaks are unisolated, so break flow continues throughout the 72-hour transient and the containment is conservatively assumed to remain at atmospheric pressure. [[

]].

The small breaks also include (1) base cases that use nominal initial conditions and nominal modeling parameters, (i.e., the PIRT multipliers set to 1.0) and (2) conservative cases that use increased initial power and increased main steam and feedwater flow, with conservative modeling parameter inputs, [[ ]]. The small breaks are modeled as single sided. Additionally, [[ ]].

The staff noted that the small steam break progresses as follows:

- A small steam pipe break occurs inside the containment concurrent with a loss of offsite power.
- Feedwater pumps trip [[ ]].
- The reactor trips [[ ]].
- Drywell high pressure is reached at [[ ]].
- [[ ]].
- The IC return valves for IC trains A and B open [[ ]].

The transient is run out to 72-hours, consistent with the analysis practice for passive emergency core cooling system (ECCS) plant applications that do not consider operator action or AC power to mitigate consequences for up to 3 days. The degradation of IC heat transfer performance is simulated by PIRT multipliers in the same manner as the large break cases.

The small liquid line pipe breaks are modeled similarly, also with two IC trains used. However, the liquid line break [[ ]].

LTR Figure 5-22 shows the break mass flow and enthalpy for the conservative case for the small steam line, and the conservative cases use the same PIRT multipliers as the large break cases. The staff also reviewed the mass flow generated by the small break cases and found them conservative in comparison to calculation of Moody results by staff at various points of time throughout the small break transient. LTR Figure 5-20 shows the RPV downcomer level and indicates that core levels are trending toward the TAF.

LTR Figure 5-27 shows the break mass flow for the conservative case for the small liquid line. The staff reviewed the mass flow generated by these cases and also found them conservative in comparison to the calculation of the Moody results by the staff at various points of time throughout the transient. LTR Figure 5-25 shows the RPV downcomer level [[

]] (LTR Figure 5-26).

The GEH CEM acceptance criterion, approved by the staff in NEDC-33910P-A, Revision 2, maintains that clad temperatures remain below initial steady-state operating temperatures. Therefore, the uncertainty involved in the heat up calculations and the margin in the predicted core water level are important parameters calculated in the method. As such, the staff focused its review on these aspects. In addition to the uncertainty in the core heat up, the uncertainties related to RPV depressurization rates are strongly correlated to the effect of radiolytic gas on IC heat transfer performance, and the staff's review also focused on these effects. The applicant indicated a design change would be made to [[
]]. Therefore, the staff considers the existing method associated with the heat transfer multiplier for IC performance to be adequate considering L&C # 1, as documented in Section 7.0 of this SE.

Relative to heat up and the reactor core water level response, the staff noted that in the core modeling, the applicant [[
]]. The TRACG code uses NRC-approved critical heat flux (CHF) correlations that have been evaluated for (GNF2 fuel) in NEDC-33005P-A, Revision 2, for small break phenomena, which is similar to CHF occurring from small break cases for the BWRX-300 design. The staff notes that the modeling used for the BWRX-300 M&E release, although more detailed than the modeling used for ESBWR, is not to the level of detail previously approved by the staff (ADAMS Accession No. ML18143A221) for BWR/2-6 ECCS-LOCA application. The staff also notes that any uncertainties used for these TRACG ECCS-LOCA analysis methods for BWR/2-6 have not been applied here, although the applicant used some of the methods to develop channel grouping and core power shape. The staff notes that it did seek clarity on whether this methodology model has been adequately quantified for phenomena where the core is forty (40) percent uncovered (i.e., collapsed liquid level) with very low decay heat. However, since this method is strictly for M&E releases and it is not the limiting case for containment design, the staff concluded that this core modeling is acceptable for this LTR based on SRP Section 6.2.1.3 and requirements of Appendix K and GDC 50. In future licensing activities where an applicant seeks to demonstrate the criteria of 10 CFR 50.46 are met for SBLOCA analyses, the staff will evaluate the safety-significance of the core heat up response involving uncovery with very low decay heat that was not encountered for the ESBWR design.

The staff found that the modeling and results of the small break steam and liquid cases are reasonable and acceptable for determining M&E releases for the BWRX-300 design considering the implementation of L&C #'s 1, 2, 3, and 4, as documented in Section 7.0 of this SE.

## 5.5 Summary of the Application Method for Large and Small Break LOCA Analyses

The BWRX-300 TRACG model shown in LTR Figures 5-3 and 5-4 describes the nodalization and the modeling used to simulate the RPV and ICS functional behavior for large and SBLOCA transients. The modeling used and inputs chosen to rely heavily on previous assessments performed for the ESBWR design. Although the vessel geometry of the BWRX-300 design is scaled down from the ESBWR design, there are several important system differences and interfaces. Some of these have been adequately incorporated into the method while others, particularly the PIRTs, do not have as clear of a basis since a BWRX-300 specific PIRT was not performed, and as such the staff focused its review in these areas.

The staff noted in its review that the large break steam and feedwater events are isolated [[ ]] that these design differences between the BWRX-300 and the ESBWR will have very minimal impact on the results. The main steam case results in the maximum M&E and long-term cooling are ample since there is minimum core inventory loss as the reactor cools up to 72-hours. Since the small break steam and liquid cases remain unisolated (i.e., open and flowing) throughout the 72-hour transient, the consequences of the event are more complex and highly dependent on adequate cooling capacity by the ICs to depressurize the RPV and maintain core inventory. With the design change [[ ]] consistent with implementation of L&C # 1 as documented in Section 7.0 of this SE, the potential for IC heat transfer degradation due to buildup of radiolytic gases becomes negligible. [[ ]]. However, the implementation of L&C # 1, 2 and 3, in Section 7.0 of the SE, ensure the method is suitably conservative, since only a very low concentration of NCGs in the IC lower drums is permitted, which eliminates gas progression into the heat transfer tubes.

The staff finds the TRACG code M&E method acceptable based on the evaluation above and because it is consistent with the guidance in SRP Section 6.2.1.3. The staff also finds the method is appropriately conservative for determining the M&E release, subject to the implementation of the L&Cs in Section 7 of this SE.

# 6.0 CONTAINMENT ANALYSIS METHOD USING GOTHIC

Section 6.0 of the LTR describes the following aspects of the GOTHIC application method and the development of the BWRX-300 containment model:

- Identification of the relevant inputs and physical phenomena relevant to the BWRX-300 containment thermal-hydraulic response.
- Description of the GOTHIC 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 GOTHIC input model for the BWRX-300 containment.

- Benchmark predictions of test data applicable to BWRX-300 containment design.

- Demonstration analyses to show the BWRX-300 containment response for various break sizes and locations using the conservative GOTHIC containment model.

- Demonstration of the capability of the PCCS to reduce the containment pressure in the long-term, recognizing the small unisolated liquid break as a potential limiting loss-of-coolant accident SBLOCA.

- 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, and the use of atmospheric pressure as the break boundary condition used to calculate conservative TRACG M&E for input to the GOTHIC containment model.

## 6.1 Generation of Thermal-Hydraulic Information for Containments Phenomenon Identification and Ranking Table

LTR Section 6.1 discusses the references and bases for identifying the phenomena important to the analysis of the BWRX-300 containment response for DBEs. The identified DBE phenomena are listed and ranked in PIRT Table 6-2 of the LTR. The purpose of the table is to assess the ability and qualification of the evaluation model for calculating the effect of the identified phenomena on the containment pressure and temperature, and to determine any additional testing, scaling or analysis needed to qualify GOTHIC for analysis of the BWRX-300 containment response. The LTR states that the initial phenomena list relevant to the BWRX-300 containment analysis was obtained by reviewing the following sources:

- NEDC-33083P-A Revision 1, TRACG Application for ESBWR.

- NEA/CSNI/R3(2014), “Containment Code Validation Matrix,” issued May 2014 (ADAMS Accession No. ML15224B463).

- SMSAB-02-02, “An Assessment of CONTAIN 2.0: A Focus on Containment Thermal Hydraulics (Including Hydrogen Distributions),” issued July 2002 (ADAMS Accession No. ML022170122).

The LTR provides further details about the selection of the phenomena applicable to the BWRX-300 containment response analysis using the above references. The staff confirmed the referenced details about the phenomena and their significance in the cited references. The applicant considered the ESBWR phenomena for applicability to DBE evaluations of the BWRX-300 design, as well as the additional ones that might apply to the BWRX-300 but not to the ESBWR. The LTR does not include the phenomena related to fan and spray dynamics or M&E exchange, as they are not applicable to BWRX-300, and does not include non-DBE phenomena.

The staff found the applicant's use of the referenced information appropriate for the development of the GOTHIC PIRT for the BWRX-300. Since the BWRX-300 has a dry containment while ESBWR has a wet containment, the staff found it appropriate that many of the ESBWR containment phenomena would not apply to the BWRX-300 containment. However, the LTR stated "the information in Reference 7.10 was reviewed for phenomena that are applicable to the BWRX-300 containment pressure and temperature analysis, including phenomena that would have equivalent phenomena in BWRX-300, even if the component was of a different design (for example, the secondary side heat transfer to the ultimate sink pools was evaluated because the phenomena are equivalent for the [[

]]. The staff concludes that considering the critical role the PCCS plays in mitigating the long-term containment pressure during an unisolated SBLOCA as well as the potential for reverse flow from the containment back into the RPV, the applicant would need to justify the PCCS secondary-side heat transfer modeling for any alternate PCCS design. This is addressed in L&C # 4 in Section 7.0 of this SE.

## 6.2 Phenomenon Identification and Ranking Table Survey

Subject matter experts retained by the applicant reviewed the PIRT, and using the criteria identified in LTR Table 6-1, ranked the phenomena in importance to the GOTHIC BWRX-300 containment pressure and temperature analysis, and identified any missing significant phenomena. To facilitate the GOTHIC method qualification for large-break loss-of-coolant accident (LBLOCA) and SBLOCA, the phenomena rankings were evaluated separately for the short-term and long-term transient evolutions. As described, in the short-term, 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. In the long-term, 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. LTR Table 6-2 summarizes the rationale used by the experts to reach the tabulated short-term and long-term PIRT rankings for the phenomena applicable to the BWRX-300 containment and LOCA scenarios.

The staff reviewed the qualitative details of the BWRX-300 containment PIRT survey methodology and found it to be rigorous and consistent with the industry best practices. Sufficient interaction took place among the experts and the applicant for the consensus building on the PIRT rankings. The phenomena identification and rankings were finalized before performing the preliminary GOTHIC calculations, which the staff considers to be conservative. The staff also confirmed that the subsequent discussion of each phenomena is consistent with the information tabulated in in LTR Table 6-2.

## 6.3 Overview of the Development of Assessment Base

RG 1.203 describes a multistep process for developing and assessing evaluation models to analyze transient nuclear power plant responses during the postulated DBEs. RG 1.203 is used to establish an acceptable evaluation method based on a well-defined application and FOM. The LTR states that the development of the assessment base follows the applicable sections of RG 1.203 guidance. It also states that most of Elements 2 and 3 in RG 1.203 have been completed as part of the GOTHIC code development and documented in the GOTHIC technical and qualification reports. The following lists the remaining items of RG 1.203 Elements 2 and 3 covered in the LTR:

- determining uncertainty in the correlations relating to the phenomena ranked high and medium based upon the existing experimental base for these correlations

- establishing suitably conservative biases in the above correlations

- establishing suitably conservative input parameters

- benchmarking the method against the integral test's representative of the BWRX-300 containment to demonstrate the conservatism in the method

The staff reviewed the information in this LTR with respect to the remaining elements of RG 1.203 for a conservative BWRX-300 containment analysis method. The BWRX-300 CEM uses the code, scaling, applicability, and uncertainty described in NUREG/CR-5249, Revision 4, "Quantifying Reactor Safety Margins: Application of Code Scaling, Applicability, and Uncertainty Evaluation Methodology to a Large-Break, Loss-of-Coolant Accident," issued December 1989 (ADAMS Accession No. ML030380503), and RG 1.203, with containment pressure and structure temperature being the two FOM. The staff noted that LTR NEDC-33911P-A, Revision 3, describes the application of RG 1.203, Element 1 to the BWRX-300 evaluation model development, up to the GOTHIC PIRT development step. The identification of the FOM, systems, components, phases, geometries, fields, and processes for the purpose of modeling had been finished as a part of LTR NEDC-33911P-A, Revision 3. The staff determined that the LTR includes the remaining information needed to develop the assessment base for BWRX-300 for using GOTHIC code and qualifications, per Elements 2 and 3 in RG 1.203.

Sections 6.1 and 6.2 of this SE discuss the staff's evaluation of the GOTHIC PIRT development and justification for their rankings. The staff considered the information provided in the GOTHIC version 8.3 code qualification documentation and found it to be adequate to address most of the PIRT as applicable to the BWRX-300 thermal-hydraulic safety analyses. The staff concludes that that the applicant appropriately identified and has included the remaining information needed to address Elements 2 and 3 of RG 1.203 that is not part of the Gothic code qualifications in the LTR to be addressed as part of RG 1.203. The staff also agrees that the BWRX-300 CEM is consistent with RG 1.203 by using a conservative analysis utilizing mature computer codes with sufficient qualification base, and is therefore, acceptable. The staff evaluation of the remainder of the information provided to address Elements 2 and 3 of RG 1.203 that is not integral to the Gothic code is provided as follows in Sections 6.4 through 6.11 of this SE.

### 6.4 <u>Knowledge Level for the Phenomena Pertinent to Containment Analysis</u>

LTR Section 6.4 describes the assessment of the available knowledge for the important PIRT phenomena that were ranked High or Medium in LTR Table 6-2. LTR Table 6-3 summarizes the knowledge level for those phenomena, as decided by the expert panel that developed the PIRT in LTR Table 6-2. The knowledge level for each phenomenon is ranked from 1 (least confidence) to 4 (most confidence) in LTR Table 6-3, which also lists the consolidated rationale for the knowledge level. The staff reviewed the rationales provided for the development of knowledge levels and agrees with the elimination of the following three phenomena from LTR Table 6-3 due to their low PIRT rankings in LTR Table 6-2:

- [[

- 
- ]]

However, the staff does not agree with the elimination of a fourth "Potential system interactions" phenomenon, due to the potential in the BWRX-300 design for reverse break flow to occur during the long-term transient evolution, and subsequent nitrogen ingestion from the containment to the RPV. Section 6.10.2 of the SE discusses the possible interaction between the containment and RPV due to potential reverse break flow and nitrogen ingestion into the ICS during a SBLOCA. This specific phenomenon will be dispositioned at the licensing stage by addressing a limitation and condition to ensure that reverse break flow does not occur, or any reverse flow that occurs is not safety-significant.

In general, the staff notes that a PIRT phenomenon ranked high in LTR Table 6-2 for importance and low in LTR Table 6-3 for knowledge level would need further investigation. Therefore, the staff found it conservative to include all remaining LTR Table 6-2 PIRT phenomena in the base GOTHIC model calculations before evaluating their modeling uncertainties, even though some of the phenomena may not be applicable or their effect may be insignificant for this BWRX-300 DBE.

## 6.5 Generation of Thermal-Hydraulic Information for Containments Model

LTR Section 6.5 presents the salient geometrical and thermal-hydraulic features of the GOTHIC model used for the BWRX-300 containment analysis. LTR Figure 6-1 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 (RCP). The LTR states that the containment dome is connected to the main containment section through two flowpaths, representing the manholes in the refueling bellows. The main cylindrical containment section and the hemispherical containment dome are nodalized in GOTHIC by using three-dimensional rectilinear sub-divided volumes, as shown in LTR Figure 6-2. [[

]].

Flowpaths are used to model the intake and exit openings of the [[ ]] connected to the RCP. These PCCS units are placed [[

]]. LTR Figures 6-2 and 6-3 show [[

]]. LTR Figure 6-3 shows [[

]]. The LTR describes how the [[

]]. LTR Figure 6-1 shows [[

]]. The LTR further describes [[

]].

The RPV is represented in the GOTHIC model by a blockage corresponding to the outer dimensions of the RPV insulation. Approximately [[ ]]of the remaining volume is assumed to be obstructed by various support structures, piping, catwalks, etc., which the staff found consistent with previous DCs. 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 M&E 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, which the staff finds to be conservative, and is specified as a function of time as obtained from the TRACG calculations. The thermal conductors for the RPV and piping are also coupled to the containment, which will account for the heat transfer from the RPV to the containment even after the break flow stops either due to the isolation valve closure or pressure equalization. The break flow path was modeled as being next to the containment shell with break flow directed toward the shell. The staff agrees that this is conservative for calculating the maximum shell temperature. The applicant submitted additional break location and flow orientation sensitivity studies to determine the limiting break location for the purpose of calculating PCP. [[ ]] is conservative for calculating the PCP, as discussed in Section 6.10.1 of the SE.

The following is a list of the additional modeling parameters and assumptions used in the GOTHIC base cases.

- [[ ]].

- The latent part of the heat transfer due to condensation is calculated using the Diffusion Layer Model (DLM) in GOTHIC (LTR Reference 7.16).

- [[ ]].

- For the shell side heat transfer coefficient calculations, the characteristic length for forced convection is set to the [[

]].

- Wall friction is calculated from the Colebrook relationship for a smooth wall.
- The form loss coefficients in the PCCS are set to conservatively high values.
- Radiation heat transfer to the shell and the PCCS is conservatively ignored.

LTR Table 6-2 recognizes several thermal-hydraulic phenomena with medium to high PIRT rankings [[

]]. As presented in LTR Section 6.5, the BWRX-300 CEM uses a [[

]].

The LTR documents that the heat transfer coefficient inside the PCCS tubes is reduced by [[ ]] in all cases, which the staff found to be conservative. The applicant also compared the McAdams correlation form used for natural convection with the Churchill and Chu's correlation that is applicable to the entire range of Rayleigh (Ra) numbers. The applicant stated that "The Nusselt (Nu) number can be calculated as the higher of the Nu numbers in the laminar and turbulent natural convection regimes instead of determining whether the flow is in the laminar or turbulent flow regimes and calculating the Nu number from the respective correlation for natural convection." The selection of the higher Nu value would capture the correct flow regime, as the flow must be laminar at the very beginning that corresponds to a higher Nu, as demonstrated by Ra~$10^{9}$, GEH provided additional information in its RAI response to RAI 06.02.01-08 dated October 8, 2021.

The staff concluded that [[

]]. Furthermore, the staff agrees that using the higher of the Nu numbers calculated for the laminar and turbulent flow in the BWRX-300 CEM for the PCCS design is also justified for both natural and forced convection regimes, as it appropriately captures the applicable convection regime.

The applicant provided an additional description in its response to RAI 06.02.01-08 dated October 8, 2021, of how the resulting density-driven single-phase flow recirculation gets established [[ ]]. The applicant provided a table showing the short-term and long-term snapshots for the PCCS parameters and nondimensional numbers (Re, Gr, Pr, Ra, and Ri) around the middle of PCCS Units #1 and #5. It also compared the small steam pipe break for PCCS Unit #1 (closest to the break) and PCCS Unit #5 (farthest from the break), for the flow velocity and temperatures of the containment space, outer and inner PCCS walls, and liquid between the inlet and middle of the PCCS. The staff determined that [[

]], as discussed in SE Sections 6.7.2 and 6.8.3. The NRC staff finds the overall GOTHIC modeling approach as discussed above to be acceptable for the BWRX-300 containment and PCCS design, based on the information provided in the LTR as well as additional information submitted in the applicant's RAI responses.

## 6.6 <u>Base Cases and Results</u>

LTR Sections 6.6.1 and 6.6.2 present the containment response to the M&E calculated in LTR Section 5.3.1 for large steam and feedwater pipe breaks using the base GOTHIC model, (i.e., without using conservative biases). LTR Table 6-4 in Section 6.6 lists the key containment inputs used in base cases, while a review of the containment nodalization and thermal-hydraulic modeling has been discussed in SE Section 6.5. As stated above, the calculations discussed in LTR Section 6.6 were performed using nominal initial conditions and nominal modeling parameters without any conservative biases and will be referred to as "base cases" that were performed using "base GOTHIC decks." The discussion in this SE will also reference "conservative cases" that were performed using "conservative GOTHIC decks," which are the same as the base GOTHIC decks except that, where appropriate, the input conditions and modeling parameters are biased in a conservative direction. The applicant submitted both base and conservative GOTHIC decks to facilitate the staff's review of the containment model dated December 17, 2020 (ADAMS Accession No. ML21028A471), then revised and updated on March 23, 2021 (ADAMS Accession No. ML21082A500) and December 14, 2021 (ADAMS Accession No. ML21348A073). The staff performed independent confirmatory analyses using the best-estimate reactor systems codes TRACE and MELCOR to analyze containment thermal hydraulic behavior in support of the staff's findings.

### 6.6.1 Containment Response to Large Steam Break, Base Case

LTR Section 6.6.1 presents information on the base case of a large steam pipe break. LTR Figure 6-4 shows the containment pressure response to a large steam line break inside the containment, with a PCP [[ ]] occurring at the time the RPV isolation valves fully close [[ ]]. The staff noted that after the closure of the RPV isolation valves, the BWRX-300 CEM appropriately accounts for the heat transfer to the containment due to convection from the RPV wall and hot pipe surfaces. With the break flow isolated, the containment pressure starts decreasing. [[

]]. As such, the staff notes that a licensing-basis analyses would need to demonstrate that the pressure in the final BWRX-300 containment design is reduced to less than 50 percent of the peak accident pressure for the most limiting LOCA within 24 hours, in accordance with the third acceptance criterion identified in of LTR NEDC-33911P-A, Revision 3, Section 3.0.

LTR Figure 6-5 shows the airspace, containment shell, and PCCS exit temperature responses. The staff found that [[

]]. The figure shows the maximum and average of all nodal air temperatures in the containment. It also plots maximum shell temperatures for the inner and outer surfaces, as well as the PCCS exit temperature. LTR Section 6.6.1 appropriately describes various physical trends captured in LTR Figure 6-5. The temperature rise in the containment is attributed to both the expansion of steam from the RPV and to the compression of nitrogen in the containment. The PCCS exit temperature shows

[[
]].

LTR Figures 6-6 and 6-7 show the decay heat and heat removal by various mechanisms following the large main steam pipe break. The staff noted that the isolation condensers are the primary mechanism for removing decay heat from the isolated RPV, [[
]].
The staff performed confirmatory analyses that also showed that various trends depicted on Figures 6-6 and 6-7 are consistent with the event progression during the large main steam pipe break.

LTR Section 6.6.1 states "No credit is taken in these cases for the heat transfer from the containment shell to the concrete supporting structures." Section 2.0 in the LTR states that "[[
]]." However, the LTR did not describe the thermal boundary condition for the containment outer surface, except for the PCCS pipes and the containment dome. To clarify the thermal boundary condition intended for the containment outer surface in the BWRX-300 CEM, the applicant described in its response to RAI 06.02.01-03, dated September 17, 2021, that the containment boundary includes a metal shell that would be free-standing (metal containment type), in loose contact with concrete (reinforced concrete containment vessel type), or in tight contact with concrete (steel concrete composite structure type). Regardless of the eventual containment type, no credit is taken for heat loss from the outer surface of the metal shell to air or concrete. This is not only an assumption used in the demonstration calculations but it is the boundary condition to be used in the application method. The applicant explained that there are also structures in the containment currently in development. The demonstration calculations did not credit the energy absorbed in these structures. The staff evaluated these conditions and agrees that neither the composition of the containment shell nor the modeling of internal structures is a limitation on the application method and finds the response to be acceptable.

LTR Section 6.6.1 also describes that the BWRX-300 containment subcompartments include the volume below the RPV, the space between the RPV and the biological shield, and the containment head area above the refueling bellows. The applicant identified the subcompartment in the containment head area above the refueling bellows as the limiting location with respect to differential pressures acting across subcompartment boundaries because of its relatively small flow area provided by the access manholes in the bellows. The other subcompartments inside the containment [[
]] have much larger openings. LTR Figure 6-8 shows the pressure differential across the refueling bellows, as calculated by the GOTHIC model using a high flow loss coefficient value [[ ]] across the manholes to maximize the differential pressure. Based on the lag between the sharp containment pressure rise closest and farthest from the break near the refueling bellows, the staff confirmed that the GOTHIC model realistically captures the pressure wave propagating through the containment immediately following the break. [[
]], which is [[ ]] of the PCP and is small for a subcompartment boundary. The staff agrees that the pressure drop across other subcompartment boundaries with larger flow area-to-volume ratios at farther locations from the break would be smaller, and that the GOTHIC model is an appropriate tool to analyze the break pressure waves and subcompartment pressurization. Determination of the jet

impingement loads acting on the containment structures is outside the scope of this LTR; therefore, an applicant or licensee will need to address the potential effect of pressure differentials on the structural integrity of the BWRX-300 containment subcompartment walls as part of future licensing actions.

### 6.6.2 Containment Response to Large Feedwater Pipe Break, Base Case

LTR Figure 6-9 shows the base case containment pressure response for a large feedwater pipe break. Similar to the large steam pipe break, the peak pressure [[ ]] is reached at the time the RPV isolation valves fully close [[ ]]. After the closure of the RPV isolation valves, the only heat input to the containment is from the RPV wall and hot pipes. [[ ]]. This demonstrates that the PCP for the large feedwater pipe break is bounded by the PCP for a large steam pipe break.

LTR Figure 6-10 shows the airspace, containment shell, and PCCS exit temperature responses for the large feedwater pipe break base case. The figure shows the maximum and average of all nodal air temperatures in the containment. It also plots the maximum shell temperatures for the inner and outer surfaces, and the PCCS exit temperature. The staff determined that all physical trends in LTR Figure 6-10 are very similar to those of the large steam pipe break captured in LTR Figure 6-5. Therefore, all the discussion in SE Section 6.6.1 of large steam pipe break also applies to the containment response to feedwater pipe break. The staff also determined that all temperature responses depicted in LTR Figure 6-5 for the large steam pipe break bound the ones in LTR Figure 6-10 for the large feedwater pipe break; and are, therefore acceptable.

.

## 6.7 <u>Nodalization Studies</u>

### 6.7.1 Nodalization Study for Containment

The LTR presents a containment nodalization study, using the base case for the large steam pipe break (LBLOCA) event discussed in SE Section 6.6.1. LTR Figure 6-11 represents the following four main containment section nodalization study cases.

- base case [[ ]]
- coarser grid with twice the node size of the base case in each direction [[ ]]
- finer grid with half the node size in each direction in the horizonal plane as the base case and the same vertical node size [[ ]]
- finer grid with half the node size in the vertical direction as the base case and the same node size in the horizontal plane [[ ]]

LTR Figure 6-12 shows the effect of nodalization on the containment pressure for the large steam break base case for each of the four nodalization schemes. The applicant chose a default [[ ]] containment nodalization as the LBLOCA base case for the PCP calculation. LTR Figure 6-12 shows a difference of [[ ]] in the calculated PCP between the base case and finer nodalization cases, which indicates [[

]] compared to the [[ ]] conservatism demonstrated in LTR Figure 6-26. The LTR states that the placement of [[ ]] was accurately done [[

]] LTR Figures 6-12 and 6-13 show little difference in the containment pressure, air/steam mixture, and shell temperatures [[ ]], which suggests that [[ ]] is adequate to resolve the phenomena controlling the containment thermal-hydraulic response. The staff found the short/long-term temperature response differences to be consistent with the pressure trend differences shown in LTR Figure 6-12, as well as the DLM's flow field modeling discussed in SE Section 6.10.1. Therefore, the staff finds the [[ ]] to be acceptable for use in the CEM for evaluation of the BWRX-300 containment response during a LBLOCA. Since the modeling uncertainties are adequately addressed (see Section 6.11 of this SE) through conservative biases and input parameters, further conservatism via the choice of nodalization is unnecessary.

The SBLOCA and LBLOCA are different DBEs that involve different phenomenological concerns. As the break flow is not isolated in a SBLOCA while it is isolated in a LBLOCA, the steam/air movement inside the containment is expected to be different. [[

]]. LTR Figures 6-12 and 6-13 show some [[ ]]. These phenomena are equally applicable to SBLOCA. With the complex SBLOCA phenomenology involving the novel PCCS design and RCP heatup in the later stage of the transient, [[ ]]. The applicant provided a similar nodalization study for limiting SBLOCA including additional information in its RAI responses and revised response to Question 06.02.01-01, dated December 17, 2021, which confirm that the relevant thermal-hydraulic phenomena were adequately captured up to 72-hours in the [[ ]] containment nodalization for the SBLOCA analysis. Section 6.10.2 of this SE contains further discussion on SBLOCA phenomena and modeling.

### 6.7.2 Nodalization Study for Passive Containment Cooling System

LTR Sections 6.7.2 and 6.8.3 present a study on the effect of nodalization on the PCCS performance. The study was conducted on a single PCCS unit placed in a large containment volume that is kept at a specified temperature and steam concentration. The PCCS and containment nodalizations in the vertical direction used in this study are the same as that of the base case BWRX-300 containment model, as described in SE Section 6.5. Like the default base nodalization, the containment has [[

]].

As the objective is to study the sensitivity of the overall PCCS heat transfer to the PCCS nodalization, the second case used in the PCCS nodalization study doubles the number of nodes both in the PCCS [[ ]] and containment, and the third case doubles the number of nodes again [[ ]]The containment pressure is [[

]], and a [[ ]] reduction is applied to the heat transfer on the outer surface of the PCCS unit. LTR Figure 6-14 shows the effect of nodalization on the heat removal rate [[

]]. The staff also reviewed additional support calculations and analysis information during the regulatory audit regarding the GEH nodalization study and performed confirmatory analyses. The staff concluded that the applicant has demonstrated that the PCCS heat transfer performance is not sensitive to the finer nodalization beyond the base case and the differences are insignificant in the context of overall containment design conservatism. SE Section 6.8.3 contains further discussion of this study.

## 6.8 Model Uncertainties and Biases

LTR Section 6.2 identifies the source of uncertainties in the phenomena important to the containment pressure and temperature response, while LTR Table 6-3 in Section 6.4 summarizes the knowledge level for these phenomena. LTR Section 6.8 presents a grouping of the phenomena and their assessment based on the observations made from the base case containment analysis results as discussed in Section 6.6. LTR Section 6.8 identifies the conservative biases needed to cover the containment modeling uncertainties based on the relevance of the potential physical phenomenon and its available knowledge level. The LTR also identifies the phenomena in LTR Table 6-3 that are inapplicable to the BWRX-300 containment modeling. The following summarizes eight observations made in the LTR in this regard.

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.

Observation 1 is physically realistic and is also reflected in various simulation results provided in the LTR. The staff accepts [[ ]] Observation 2 to be a bounding assumption. The staff finds [[ ]] Observation 3 to be conservative [[ ]]. The staff also accepts [[ ]] GOTHIC inputs, to address several PIRT items. As Observations 1-4 are based on conservative assumptions, no quantification of their uncertainties is warranted.

Observation 5 pertains to the natural and forced circulation and stratification that would be driven by the friction factors and turbulence and could be sensitive to the model nodalization. LTR Section 6.8.1 presents a sensitivity study for the friction factors and turbulence modeling that shows [[ ]], while LTR Section 6.7 presents a nodalization study for the containment and PCCS. LTR Section 6.8.3 presents a sensitivity study for the PCCS performance closely tied to Observation 5 [[ ]]. LTR Section 6.8.2 adequately develops bounding values of uncertainties in the convection and condensation heat transfer coefficients, to address Observation 6. Observation 7 is addressed via L&C #1 documented in Section 7.0 of this SE, by limiting the amount of radiolytic gases to avoid ICS performance deterioration and hydrogen deflagration. The staff confirmed that the multi-component gas mixture properties in Observation 8 are calculated within GOTHIC as described in GOTHIC Thermal Analysis Package Technical Manual, Version 8.3(QA), issued in 2019 by the Electric Power Research Institute.

### 6.8.1 Effect of the Friction Factors and Turbulence Parameters on the Containment Response

LTR Section 6.8.1 presents a sensitivity study of the effect of friction factor and turbulence parameters on the containment pressure and temperature response. The study started from the nominal cases that use Colebrook's friction factor for a smooth surface, [[ ]]. A relative roughness of [[ ]] was used to study the sensitivity of the containment pressure and temperatures to the friction factor, which corresponds to a very high absolute value of surface roughness due the large containment hydraulic diameter. The LTR presents a case [[ ]] ("High Cf" case). Likewise, two cases of [[ ]] (high $C_{\mu}$) and [[ ]] (low $C_{\mu}$) [[ ]].

LTR Figures 6-15 through 6-17 compare the results obtained by the above sensitivity cases for the containment pressure/temperature responses and steam stratification for the large steam pipe break. The pressure and temperature plots for the sensitivity and nominal cases in LTR Figures 6-15 and 6-16 [[ ]]. During the regulatory audit, the staff reviewed details of the sensitivity studies and confirmed that they were appropriate to demonstrate [[ ]]. The staff did not explore the sensitivity for the SBLOCA, because the uncertainties quantified in condensation and convection heat transfer modeling, as discussed in Section 6.8.2 of this SE, are expected to bound any uncertainties in modeling turbulence and friction characteristics for the containment model. Therefore, the staff determined [[ ]]. A similar sensitivity study was performed for the PCCS separately, and is discussed in Section 6.8.3 and of this SE.

### 6.8.2 Uncertainties in the Convection and Condensation Heat Transfer Coefficient and the Bounding Values

Section 6.9 of this SE presents the staff evaluation of the benchmarking of the GOTHIC code containment simulation against the Carolinas Virginia Tube Reactor (CVTR) integral test data, which justified use of the conservative DLM condensation option without the film enhancement feature, as built into GOTHIC, for the BWRX-300 containment analysis. The additional biases included to the DLM model to bound the experimental uncertainties in the convection and condensation heat transfer correlations used in the BWRX-300 containment GOTHIC model are further discussed below.

The DLM condensation option used by the applicant is a mechanistic model which represents the underlying phenomena with a heat and mass transfer analogy. The total heat transfer to a condensing surface has two parts, the convection heat transfer from the bulk gas to the condensing liquid film and the heat transfer across the film thickness. The LTR presents a benchmarking of the similar Heat and Mass Transfer Analogy Method (HMTAM) against the CONdensation with Aerosols and Non-condensable gases (CONAN and COPAIN facility) test data to quantify the uncertainties in the convection and condensation correlations. The LTR also presents information to show that the DLM method used in the BWRX-300 CEM is more conservative than the HMTAM, which justifies use of the experimental uncertainties determined using the HMTAM in the BWRX-300 GOTHIC model.

LTR Figure 6-18 compares the forced and natural convection correlations to the COPAIN test data. The figure shows the ratios of the measured Nu number to the NU numbers calculated by using the Schlichting forced convection correlation and the McAdams natural convection correlation, for the entire range of the test data. By predicting each measured data point with both the forced and natural convection correlations, the applicant identified the applicable Richardson number ($Ri = Gr/Re^2$) ranges for the forced, natural, and mixed-convection regimes. The Richardson number represents the significance of the buoyancy forces with respect to inertial forces. LTR Figure 6-18 identifies the respective Richardson number regimes: [[

]]. LTR Figure 6-18 clearly links the transition region with the maximum nonconservatism in heat transfer predictions, (i.e., where the forced and natural convection correlations overpredict the measured Nu number the most, thus, leading to the highest uncertainty in the test data). The LTR also accounts for the "relaminarization" of flow near the wall that would reduce the heat transfer by suppressing the turbulence in the natural convection flow near the wall until the free stream velocity is large enough to overcome the buoyancy forces. On the vertical containment walls at temperatures lower than that of the containment airspace, relaminarization would take place only when the bulk circulation flow is downward. LTR Figure 6-18 shows that this occurs in the transition region between natural and forced convection [[ ]].

For condensation, the correlation forms and coefficients for forced and natural convection remain the same as those used for convection, but the Prandtl (Pr) number is replaced by the Schmidt (Sc) number, and the Nu number is replaced by the Sherwood (Sh) number, and all the above discussion is equally applicable. LTR Figure 6-20 shows that

]]. Based on these considerations, LTR Section 6.8.2 proposed the following convection and condensation correlation biases used in the GOTHIC model of the BWRX-300 containment to ensure that the heat transfer coefficients are appropriately reduced to bound the test data:

- convection correlation bias:

  1. [[

  2. ]]

- condensation correlation bias:

  1. [[

  2. ]]

The LTR clarifies that [[

]].

In order to disposition the possibility that GOTHIC may incorrectly predict the flow-direction or convection-mode and nonconservatively apply lower biases to the heat transfer coefficient, the applicant provided information on GOTHIC's qualifications to accurately predict the flow field (magnitude and direction) in the near wall region. The applicant also provided a summary of several GOTHIC validations against measured velocity data for situations that involve multidimensional flows where buoyancy forces are significant.

To quantitatively assess the use of the flow-direction/convection-mode dependent conservatisms used in the BWRX-300 CEM, the applicant performed a 180 degree break flow orientation sensitivity study in its response to RAI 06.02.01-03 dated September 17, 2021, and further clarified in RAI response to Question 06.02.01-01,dated October 29, 2021. This study modeled the break flow as being directed upward, downward, and sideways toward the containment shell. It was performed for the most limiting break location with respect to PCP, (i.e., [[

]], the applicant did not perform a small break orientation sensitivity study, citing that a small break case would show [[

]]. The staff reviewed the details of the sensitivity study as part of an audit and confirmed that the conclusions would be applicable to the range of conditions expected for potential applications of the CEM.

The staff also recognizes that benchmarking of the test data discussed above for the HMTAM model of condensation uses the Schlichting correlation for forced convection, while the BWRX-300 containment model uses [[

]]. LTR Figure 6-21 shows [[

]].

Based on the above evaluation, the staff concludes that the applicant has demonstrated GOTHIC's capability to calculate the velocity field in a vessel resulting from jets and buoyancy, and that the applicant has justified the use of the specified flow-direction/convection-mode dependent biases in the condensation/convection correlations. Therefore, the applicant's use of the conservative DLM condensation option [[ ]], as described in the LTR, is acceptable for the BWRX-300 containment analysis.

### 6.8.3 Sensitivity Analyses for PCCS Performance

LTR Section 6.8.3 presents a sensitivity study of the effect of the following parameters on the PCCS performance.

- PCCS loss coefficients
- PCCS liquid-side heat transfer coefficient
- fouling

The study was conducted on a single PCCS unit placed in a large containment volume that is kept at the specified containment pressure, temperature and steam volume fraction values, while each of the sensitivity parameters was individually varied to determine its effect on the PCCS performance. The base case PCCS and containment nodalizations in the vertical direction are the same as that used in the BWRX-300 containment model described in LTR Section 6.5, with the exception of using one node in the horizontal direction. The base case has no fouling or paint. The staff found using a single node in the horizontal direction appropriate, as the objective of this standalone PCCS study was to understand the sensitivity of the steady-state PCCS heat removal rate to specific parameters beyond the condensation heat transfer coefficient discussed in Section 6.8.2 of this SE. The applicant conducted the sensitivity study for each of the following conditions.

- containment pressure: [[ ]]
- containment airspace temperature: [[ ]]
- steam volume fraction: [[ ]]
- PCCS inlet temperature: [[ ]]
- PCCS total loss coefficient: [[ ]]

Reviewing LTR Section 6.7.2, the staff determined that the above conditions are steady-state snapshots of the containment at different times during the analyzed DBE. LTR Table 6-5 summarizes the results of the PCCS sensitivity cases. [[

]]. The LTR

documents the thermal conductivities of the paint and crud on the outer and inner surfaces of the PCCS, respectively.

The staff in its regulatory audit reviewed the supporting GOTHIC decks and detailed PCCS modeling information and performed confirmatory calculations. The applicant explained that [[

]]. The confirmatory calculations performed by the staff confirmed that [[

]]. Furthermore, the confirmatory calculation results verify the applicant's characterization [[

]].

The staff also notes that use of the CEM was demonstrated for the BWRX-300 design with the specific [[ ]] configuration and channel placement described in this LTR. The BWRX-300 safety analyses as presented in the LTR, and the additional sensitivity studies, are based on the premise that [[ ]]. For example, the applicant did not perform an azimuthal study of the break location as a part of the CEM [[ ]]. However, the LTR mentions that the PCCS design has not been finalized, which means that the PCCS design submitted at the licensing stage may not have the same [[ ]]. An alternate PCCS design may involve additional phenomena not reviewed as a part of BWRX-300 method. The applicant did not provide sufficient information to generically extend the results from the GOTHIC sensitivity studies and justifications for the applicability of the CEM to other proposed PCCS designs. Therefore, the staff has imposed a L&C #4 as documented in Section 7.0 of this SE, to ensure that the BWRX-300 CEM is evaluated for its capability to accurately analyze any proposed alternative PCCS design configuration and layout at the licensing stage.

## 6.9 Benchmarking to the Carolinas Virginia Tube Reactor Integral Tests

LTR Section 6.9 provides a high-level overview of benchmarking the GOTHIC code against the CVTR integral test data. The applicant has referenced the CVTR integral tests to benchmark the GOTHIC code simulation against the CVTR test data for a steam pipe break with slightly superheated steam injection into a closed containment. LTR Figure 6-22 illustrates the CVTR Test Facility geometry, and the corresponding three-dimensional GOTHIC model used for the qualification. The applicant used Test Case #3 as the most applicable to the BWRX-300 containment [[ ]]. The staff found the selection of CVTR containment geometry and Test Case #3 for BWRX-300 application acceptable because [[

]]. The staff confirmed the corresponding descriptions of the test facility and qualification of the GOTHIC model in the respective references.

LTR Figures 6-23, 6-24, and 6-25 show the CVTR test data benchmarking for the containment pressure, airspace temperature (thermal stratification), and structure temperature, respectively. In the CVTR assessment, a condensation heat transfer sensitivity study was performed to investigate the pressurization effect, which showed that [[

]]. The three-dimensional GOTHIC model using the DLM-FM model with condensation heat transfer enhancement due to film roughening predicts the measured pressure, thermal stratification, and structure temperatures closely. During the audit, the applicant confirmed [[

]] in LTR Figure 6-23. LTR Figure 6-23 demonstrates that the application of the conservative DLM option available in GOTHIC (DLM condensation option without the film enhancement feature), along with the condensation/convection heat transfer biases described in Section 6.8.2, leads to a bounding prediction of the CVTR test data. The staff confirmed through the audit that the red curve in LTR Figure 6-23 (DLM, Conservative) was generated by the applicant using an appropriate GOTHIC model developed for the CVTR facility.

Applying the same CVTR GOTHIC model, the applicant also generated curves for the corresponding case using the Uchida correlation, which is a well-supported condensation correlation in the literature. The comparisons in LTR Figures 6-23, 6-24, and 6-25 demonstrate that the containment pressure, airspace temperature, and structure temperature predicted using the conservative DLM option with biases bound the values predicted by the best-estimate DLM option, Uchida correlation, as well as the CVTR test data. Code limitations prevented the applicant from generating the Uchida correlation curves using all conservative biases from LTR Section 6.8.2, however, this does not affect the essential conclusion of this benchmarking study, that the conservative DLM option selected for the CEM bounds the test data.

LTR Figures 6-24 and 6-25 compare the CVTR GOTHIC model predictions against the test data for the airspace and structure surface temperatures, respectively. The calculations were performed for the test data at two airspace and structure elevations, using conservative DLM, Uchida, and best-estimate heat transfer models. The staff noted that the predicted airspace and structure surface temperatures are higher than the measured temperatures and capture the temperature stratification trends observed in the CVTR test data well before and after the steam flow stops. However, the staff confirmed that the predicted airspace and structure surface temperature differentials for the biological shield elevation in LTR Figures 6-24 and 6-25 are close to the measured CVTR temperature differentials. Furthermore, the differences in the calculated temperatures in LTR Figures 6-24 and 6-25 at higher and lower elevations are close to the differences in the measured temperatures. Therefore, even though the CVTR GOTHIC model predictions are biased toward a higher airspace temperature, they would not result in a higher condensation rate due to an equally biased higher surface temperature. The staff also noted that the conservatism inherent in the GOTHIC conservative DLM model is manifested by the fact that it typically bounds the Uchida and best-estimate models.

Based on the information provided in the LTR about the CVTR benchmarking, the staff agreed that the conservative DLM condensation option when used with additional condensation/convection heat transfer biases appropriately bounds the CVTR test data for pressure and temperature stratification trends. Therefore, this modeling approach is acceptable for use in the BWRX-300 CEM that resolves three-dimensional effects via GOTHIC nodalization.

## 6.10 <u>Demonstration of the Method for Large and Small Breaks, Conservative Cases</u>

LTR Sections 6.10.1 and 6.10.2 provide information to demonstrate the conservatism in the evaluation method for containment response to the large and small steam pipe breaks. For that purpose, the LTR presents the containment response results for the conservative cases for the limiting large and small steam pipe breaks, as discussed below.

### 6.10.1 Containment Response to Large Steam Pipe Break, Conservative Case

LTR Figures 6-26 and 6-27 compare the containment pressure and temperature responses for a large steam pipe break LBLOCA case, calculated using the conservative case assumptions and the base case assumptions. LTR Figure 6-26 shows the calculated PCP to be [[

]]. LTR Figure 6-26 also shows that containment gauge pressure decreases to [[ ]], continuing to follow a decreasing trend for both the base and conservative cases. The staff evaluated these assumptions and concludes that the biases used in the CEM add a significant conservatism to the peak pressure values, which the staff finds to be sufficient to address the uncertainties in the method.

For the final BWRX-300 design, the applicant would need to demonstrate that the acceptance criteria in SRP Section 6.2.1.1.A, “PWR Dry Containments, Including Subatmospheric Containments,” Revision 3, issued March 2007 (ADAMS Accession No. ML063600402) are met for the limiting design-basis accidents: (1) the containment design pressure should provide at least a 10-percent margin above the calculated conservative PCP, and (2) the containment pressure should be reduced to less than 50 percent of the peak calculated pressure within 24 hours after the postulated accident.

LTR Figure 6-27 shows the calculated maximum shell temperature to be [[ ]] for the conservative LBLOCA case. The NRC staff found that the trends in this figure correlate to those in the break flow and enthalpy from TRACG, as well as the biases included in the conservative case. The staff determined that they are also consistent with the trends demonstrated by the staff confirmatory analysis.

LTR Figures 6-28, 6-29, and 6-30 compare the containment pressures, heat transfer rates, and containment temperatures results predicted by the biased DLM and Uchida condensation correlations. All comparisons were made between the biased DLM and Uchida condensation correlation predictions for the conservative large steam pipe break case, with all other inputs and assumptions the same. LTR Figure 6-28 shows that the difference in the predicted PCPs is negligible. The staff agrees that the biased DLM correlation predicts higher shell temperatures in LTR Figure 6-30 due to its ability to account for the impact of break velocities on condensation heat transfer. As the Uchida correlation does not account for the flow field, it underpredicts the shell temperature in the presence of break flow in the near wall region. The staff concludes that after the initial break flow impact subsides, the Uchida correlation calculates a higher condensation rate on the shell and PCCS and, thus, leads to less conservative long-term containment pressure and temperature responses, (i.e., a faster reduction in containment pressure). The staff finds that the applicant has demonstrated a sufficient level of conservatism in the biased DLM correlation used in the CEM and its ability to account for the flow field effect on condensation heat transfer compared to the Uchida correlation, as also demonstrated in the CVTR benchmarking discussed in Section 6.9 of this SE.

LTR Figure 6-11 presents the vertical and horizontal cross-sectional views of the three-dimensional GOTHIC grids used for the BWRX-300 containment model. It shows that [[

]]. The analyses presented in the LTR [[

]]. To address this possibility, the applicant performed a break location sensitivity study in its response to RAI 06.02.01-03 dated September 17, 2021, and further clarified in RAI response dated October 29, 2021, for the conservative case of large steam pipe break LBLOCA, and presented the description and results for the following three break locations, with the break flow directed toward the containment wall:

(1) placed near the containment wall [[ ]], which is the same break location and orientation as presented in LTR Section 6.10.1

(2) placed near the RPV [[ ]], which is at the same horizontal level as (1) but is located radially inward closer to the RPV

(3) placed near the RPV [[ ]], which would be [[ ]] below (1) and (2)

The applicant provided figures comparing the containment pressure and temperature responses for the initial four hours, for the three large break location cases. [[

]]. The staff finds this treatment of the limiting break locations for calculating the PCP and maximum shell temperature to be acceptable because it maximizes the conservatisms for the two FOM under DBA conditions.

### 6.10.2 Containment Response to Small Steam Pipe Break, Conservative Case

Limiting Small-Break Loss-of-Coolant Accident

The applicant provided conservative case GOTHIC analysis results for a small steam pipe break and a small liquid pipe break. A comparison showed that the results for the steam and liquid SBLOCA are similar, and [[

]]. The conservative case containment response for small pipe breaks is shown in LTR Figures 6-31 through 6-34 for steam SBLOCA and in LTR Figures 6-39 through 6-41 for liquid SBLOCA. Based on the LTR description, the staff understands that, for the BWRX-300 licensing basis, both steam and liquid SBLOCA scenarios will be evaluated for the PCP and maximum shell temperature at the respective limiting location and break flow orientation.

The applicant also performed a break location and flow orientation sensitivity case for a steam SBLOCA in its response to RAI 06.02.01-03 dated September 17, 2021, and further clarified in RAI Question 06.02.01-01 response dated October 29, 2021, at the limiting PCP and maximum shell temperature break locations discussed for the LBLOCA in SE Section 6.10.1. The results provided in the response for 72-hours show that [[ ]]. Based on the results, the limiting SBLOCA [[ ]]. The maximum containment pressures for SBLOCA [[ ]] are significantly lower than the LBLOCA PCP [[ ]], while the maximum shell temperatures are somewhat comparable for SBLOCA ]] and the limiting LBLOCA [[ ]].

Small-Break Loss-of-Coolant Accident Nodalization Study and Containment Backpressure

The applicant performed a SBLOCA nodalization study to show that the default base case nodalization [[ ]] would be acceptable to predict the SBLOCA FOM for the BWRX-300 containment. LTR Figures 6-31 and 6-39 show that the containment pressure responses for the default [[ ]] containment grid with conservative inputs and model biases are similar for the steam and liquid SBLOCAs. The TRACG model in the BWRX-300 CEM does not credit the containment back pressure in calculating the small break M&E into the containment for its response calculations. This is a conservative assumption because, if containment back pressure was realistically considered, the break flow would become unchoked and start decreasing when the pressure differential between the RPV and the containment becomes low enough. Not crediting containment back pressure in break flow calculations also has the convenience of avoiding iteration and convergence between the TRACG and GOTHIC solutions.

The applicant also states that the RPV and containment pressures would eventually equalize, after which there is essentially no more mass discharged to the containment unless the containment pressure decreases at a faster rate than the RPV pressure. LTR Figure 6-31 shows a simplified analysis of the pressure equalization. Assuming a constant atmospheric pressure boundary condition in the BWRX-300 CEM until the RPV-containment pressure equalization point in LTR Figure 6-31 maximizes the M&E into the containment and hence, its pressure. The blue-dashed RPV curve in LTR Figure 6-31 represents the highest possible RPV pressure [[ ]]. The applicant used the limiting [[ ]] RPV pressure as the upper bound for the containment pressure for break cases, as shown by the red dashed extension of the blue-dashed curve in LTR Figure 6-31. The staff finds the [[ ]] RPV pressure as the upper bound on containment pressure for equalization to be reasonable, as it would be higher than the lower RPV pressure bound [[ ]], as shown by the solid blue curve in LTR Figure 6-38.

The lower containment pressure bound shown by the red dotted curve in LTR Figure 6-31 is calculated by assuming no more break flow in the SBLOCA after the time the RPV and containment pressures equalize. This approach to establishing the upper and lower bounds means that the upper containment pressure bound is dictated by the ICS heat removal capacity, while the lower containment pressure bound is dictated by the PCCS heat removal capacity. A conservative prediction for the containment pressure is the solid red curve calculated using atmospheric pressure as the containment backpressure for the M&E release until the point of pressure equalization, followed by a value that would be between the upper and lower bounds. In response to RAI 06.02.01-01, dated December 17, 2021, the applicant provided an expanded LTR Figure 6-31 which includes all four nodalization schemes discussed in LTR Section 6.7.1, as well as a lower bound on the RPV pressure with SBLOCA break flow accounted for, while the upper bound is still established by the RPV pressure [[ ]]. The staff recognizes that the lower and upper bounds of containment and RPV pressures are introduced mainly to illustrate that the general containment pressure response and pressure reduction trend are not safety-significant for all four containment nodalizations.

LTR Figure 6-32 shows the PCCS exit temperature and the RCP temperature. The staff concludes that the [[ ]]. LTR Figure 6-33 shows the maximum and average steam volume fractions in the main containment. LTR Figures 6-31, 6-32, 6-33, and 6-38 reflect that [[ ]]. The LTR states that the calculations conservatively assume no heat loss from the RCP to the surroundings through the walls, but they do account for the heat loss due to surface evaporation from the pool. LTR Figure 6-34 shows that the peak shell temperature trends are similar for SBLOCA and LBLOCA with the expected time lag.

As discussed earlier, the break flow is maximized by assuming the containment to be at atmospheric pressure until the pressure equalization point. However, if the ICS can depressurize the RPV faster than the PCCS can depressurize the containment, NCGs, such as nitrogen may get ingested into the RPV and accumulate in the ICS heat exchangers. This may degrade the heat removal rate of ICS, causing the system to repressurize again. The applicant investigated the NCGs ingestion into the RPV by performing a coupled TRACG-GOTHIC calculation for a steam SBLOCA by accounting for the effect of containment back pressure on unchoked break flow using TRACG-GOTHIC iterations. LTR Figures 6-36, 6-37, and 6-38 show the results of the coupled TRACG-GOTHIC calculations. LTR Figure 6-36 shows that the RPV and containment pressures do not equalize, and the pressure differential between the RPV and containment becomes smaller with time, [[ ]]. LTR Figure 6-37 confirms a positive break flow and, thus, no break flow reversal. LTR Figure 6-38 is very similar to LTR Figure 6-31 as discussed above, but it also shows the coupled TRACG-GOTHIC calculation results from iteratively using the containment pressure calculated by GOTHIC as back pressure in calculating the break flow by TRACG. [[ ]] in LTR Figure 6-38, [[ ]]. The red dashed line in LTR Figure 6-38 also shows the containment pressure resulting from the break flow calculated based on a constant atmospheric pressure boundary condition for the TRACG M&E release calculations.

The staff evaluated the results presented in LTR Figure 6-38 and concludes that ignoring the back pressure in TRACG M&E release calculations introduces conservatism in the long-term GOTHIC containment response for the small break cases. LTR Figures 6-39 through 6-42 show the containment responses for the small liquid pipe breaks until the RPV and containment pressures equalize, [[ ]]. The staff agrees that the containment response trends for the small liquid and small steam pipe breaks are similar. The staff agrees that [[ ]], the liquid SBLOCA behaves like a steam SBLOCA.

Based on the above discussions, the staff concludes that the LBLOCA and SBLOCA nodalization studies provided by the applicant have demonstrated that the nodalization selection does not significantly impact the results for the FOM, (i.e., the PCP resulting from LBLOCA and the containment depressurization to less than 50 percent of the limiting PCP value within 24 hours). Therefore, the default base case nodalization [[ ]] using the CEM is an acceptable representation of the BWRX-300 for containment safety analyses.

The staff did not review the coupled TRACG-GOTHIC calculations as a part of the process by which the PCP and maximum shell temperature are determined, since the LTR clearly states that no containment back pressure is credited in the calculation of the FOM. The staff finds the qualitative trends predicted by the coupled calculations by crediting the back pressure for the example case to be realistic and considers this to be a reasonable approach to demonstrating that flow reversal does not occur through the first 72-hours of a SBLOCA. The staff cannot generically extend the conclusion from the study to the future final design for BWRX-300; therefore, the staff has formulated L&C # 3 as documented in Section 7.0 of this SE that would require a demonstration at the licensing stage of no safety-significant break flow reversal during the first 72-hours of the event.

<u>TRACG/GOTHIC Model Changes</u>

The applicant submitted updated information supporting the LTR review that included the original and updated sets of TRACG and GOTHIC models for the LBLOCA and small steam break LOCA base and conservative cases. The results presented in LTR Section 6.10 were generated using the last updated models from December 14, 2021, with some additional changes resulting from RAI responses, but the sensitivity studies were performed using the original models and were not repeated with the latest models. The applicant stated that the results in LTR Section 6.10 showed small differences relative to those in LTR Revision 0 due to modeling differences and use of a different code version. The staff compared the LTR Section 6.10 results from LTR Revision 0 against those from LTR Revision 2 and agrees that the updated results demonstrate that no new or additional phenomena were introduced by the model/code changes. As these differences affect all base and conservative cases in the same manner, they do not affect any of the uncertainty discussions. Therefore, the staff confirmed that no major CEM conclusions based on the original models regarding the limiting transients, rapid cooling requirements, nodalizations, and modeling uncertainties have been adversely affected by the updated models.

### 6.10.3 Containment Mixing for Combustible Gases

LTR Section 5.2.4 presents the discussion and calculations for the radiolytic hydrogen and oxygen generation in BWRX-300 DBAs. Oxygen and hydrogen are generated from radiolysis in a stoichiometric ratio of 0.5. As the radiolytic gases are well mixed in steam or liquid water, the volumetric ratio of radiolytic oxygen to hydrogen remains 0.5 as they migrate in the RPV, from the RPV to the containment, and within the containment. The radiolytic gases mixed in steam and liquid are discharged into the containment along with the break flow, and there is also a small amount of oxygen is also present initially in the containment. Radiolytic hydrogen and oxygen distributions in the containment are not a concern for large breaks [[

]]. However, radiolytic gases may build up in the containment following unisolated small breaks over time even though the rate of release is very small. LTR Figure 6-43 shows the radiolytic gas generation and release from the RPV for the limiting conservative case of a small steam pipe break with two ICS trains. It shows that the total amount of radiolytic hydrogen and oxygen produced in the RPV during the first 72-hours is [[ ]].

The radiolytic gas volume fractions are specified in the break flow boundary condition as calculated in LTR Section 5.2.4, and the radiolytic gas volume fraction distribution in the containment and the dome region was calculated for the limiting conservative case of a small steam pipe break with 2 ICS trains. LTR Figure 6-44 shows the average and maximum hydrogen volume fractions in the main containment volume and the dome region. [[

]].

Although the volume fraction of radiolytic gases in the steam is small, it may accumulate in the dome region over time. The applicant clarified that the deflagration and detonation limits for the combustible gases depend not only on the hydrogen and oxygen volume fractions, but also on other parameters, such as pressure, temperature, steam volume fraction, potential flame propagation directions, and geometry. The flammability limit for hydrogen in dry air is 4 percent, and the minimum oxygen required for combustion to take place is 5 percent. The autoignition temperature, whether it is the gas temperature or the surface temperature, is in the range of 580–800°C. The flammability limits of hydrogen documented in two referenced cases of different conditions in containment are 7.3-7.9 percent, and 8 percent respectively. When the steam fraction is 40 percent, the required hydrogen fraction in the mixture increases to as much as 15 to 40 percent. Combustion is inhibited at even higher steam concentrations. In all the referenced cases described in the response, the flammable limits are well above 4 percent. The deflagration and detonation limits are above the flammable limit.

The results in LTR Figure 6-44 show that the 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 differences in maximum and average combustible gases in the main containment and containment dome are very small. It indicates that sufficient mixing of combustible gases has been demonstrated. In addition, the LTR Section 1.2 states that the BWRX-300 containment subcompartments are well mixed due to the open connections between containment and the volume below the RPV and containment and the space between the RPV and the biological shield. Sufficient mixing in containment subcompartments eliminates the concern for deflagration in the containment subcompartments. In the response to RAI 06.02.05-01 dated September 17, 2021, GEH clarified that the hydrogen volume fractions in the containment dome shown in LTR Figure 6-44 for BWRX-300 are far below the deflagration limits by 2 orders of magnitude even if there is sufficient oxygen. Based on the additional information, the staff concludes that the calculated combustible gases concentrations inside the containment following unisolated small breaks are far below the deflagration limits.

Based on above, the NRC staff confirms that gases generated by radiolysis are not likely a concern for deflagration in the containment and the identified containment subcompartments could have sufficient mixing, contingent on the final design. The NRC staff will review the specific features of the containment design during future licensing activities for the BWRX-300 SMR.

### 6.11 Summary of the Assumptions and Inputs Used in the BWRX-300 GOTHIC Method Conservative Cases

LTR Section 6.11 summarizes the assumptions and inputs to be used for the conservative BWRX-300 GOTHIC containment analyses as part of the NEDC-33922P CEM.

Both the base and conservative cases for containment analyses use the following modeling parameters:

- The initial containment pressure is at the maximum technical specification containment pressure.
- The initial bulk containment temperature and the structures are at a reasonably low value that would occur during normal operation.
- The initial humidity of the containment airspace is 20 percent.
- The initial RCP temperature is at the technical specification limit. For the purpose of demonstration calculations, [[ ]].
- Bounding values are used for form loss coefficients in the containment and in the PCCS units.
- The free space volume in the containment is conservatively calculated.
- The containment is modeled using a [[ ]] nodalization, as presented in the base cases in the LTR.
- The initial airspace temperature above the RCP is assumed to be the same as the pool water temperature.
- The initial relative humidity of the RCP airspace is assumed to be 100 percent.

The conservative cases for containment analyses use the following biased modeling parameters and conservative modeling assumptions:

- [[ ]]
- The condensation and convection heat transfer correlations are biased as described in Section 6.8.2. The resulting heat transfer coefficients on the containment shell, PCCS and containment dome are appropriately reduced.
- [[ ]]
- [[ ]]

- No credit is taken for heat transfer from the outer surface of the metal containment shell to the concrete or surroundings, except for heat transfer from the submerged section of the containment dome to the RCP above the dome.

- The RCP is modeled as a lumped parameter volume, whose air space is connected the ambient atmosphere through a constant pressure boundary condition, such that the airspace pressure remains nearly constant.

- There is no heat loss from the RCP to the walls. Surface evaporation from the pool is accounted for.

The applicant also provided additional information in LTR Table 6-2 regarding the following PIRT phenomena that are pertinent to the RCP for the first 24 hours of a LBLOCA and first 72-hours of a SBLOCA.

- [[

- ]]

Based on the information provided by the applicant, the staff found that modeling the airspace above the RCP at an initial temperature equal to the uniform pool water temperature, and an initial relative humidity of 100 percent, is conservative. The staff found the RCP liquid modeling at a uniform temperature with no stratification, to be conservative as PCCS intake location in the pool is near the bottom of the pool, and the PCCS return pipe discharge elevation is at least [[ ]] the PCCS intake elevation. The remainder of the assumptions listed above are consistent with the information discussed and found to be acceptable earlier in this SE. The staff also found the information to be consistent with the submitted GOTHIC models, and thus, these assumptions capture the staff findings based on the sensitivity studies, demonstration calculations, and confirmatory analyses.

The applicant additionally clarified that the applicability of the BWRX-300 CEM is not based on a time limit, such that as the first 72-hours of the DBE, but on the modeled phenomena and biases. The PCCS heat removal capacity will decrease as the RCP heats up, but the PCCS heat transfer modeling does not have any specific limitation until boiling starts in the PCCS tubes. The applicant recognized that the capability of the GOTHIC model has not been demonstrated for specific phenomena such as the release into the containment of significant amounts of hydrogen or superheated steam from the RPV. The staff finds this justification to be acceptable as these phenomena do not occur within 72-hours during a DBE
the scope of this LTR does not include approval for the quantification of their uncertainties.

The conservatism in the BWRX-300 CEM evaluation model is achieved by simultaneously biasing the individual inputs and key modeling parameters to bound the uncertainties in the conservative cases, rather than performing a statistical sampling of the uncertainties and adding margin to the base case results. The staff agrees that compounding conservatisms through individual biases greatly reduces the number of sensitivity runs needed for statistical sampling, and gives reasonable assurance that the overall analysis results bound the uncertainties.

## 7.0 LIMITATIONS AND CONDITIONS

If an applicant chooses to incorporate the LTR by reference as part of an application for approval of a reactor design, or a construction or operating license, it must abide by the following L&Cs or provide additional justification for any deviations.

### 7.1 L&C #1 - Isolation Condenser Radiolytic Gas Removal

The use of this CEM is limited to a BWRX-300 design that limits the total volumetric fraction of radiolytic gases in the IC lower drum to a sufficiently low level throughout a 72-hour period following the event such that condensation heat transfer in the ICs is not adversely affected and the hydrogen deflagration margin is maintained.

### 7.2 L&C #2 - Isolation Condenser Return Line Design and Further Demonstration of Transient Reactor Analysis Code General Electric Modeling Capability

The use of this CEM is limited to a BWRX-300 design that a proper isolation condenser return line layout is chosen, such as a loop seal or a water trap, to prevents reverse flow from RPV into the IC return line throughout a 72-hour period following the event or where an applicant or licensee referencing this report demonstrates that the TRACG code is capable of conservatively modeling the overall ICs heat removal capacity when reverse flow occurs in the IC discharge lines.

### 7.3 L&C #3 – Demonstration of No Safety-Significant Break Flow Reversal During the First 72-Hours into the Event

The use of this CEM is limited to a BWRX-300 design in which the PCCS is sized sufficiently large such that a reverse flow from containment back to RPV does not occur during the first 72-hours into the event. The applicant or licensee referencing this report needs to demonstrate that no reverse flow could occur, or any reverse flow that occurs under the most bounding flow reversal conditions resulting in the degradation of IC heat transfer is not safety-significant with respect to the acceptance criteria for the BWRX-300 CEM.

### 7.4 L&C #4 – Demonstration of the Applicability of the BWRX-300 Containment Evaluation Method to the Final Passive Containment Cooling System Licensing-Basis Analysis

The use of this CEM was demonstrated for a BWRX-300 design with the [[
]] and placement described in this LTR. For any alternate PCCS design configuration and placement, the applicability of this method and the PCCS modeling approach must be reviewed and found to be acceptable by the NRC for BWRX-300 licensing-basis analyses.

## 8.0 CONCLUSIONS

The GEH LTR provides sufficient information to justify the method to evaluate the BWRX-300 containment response for the acceptance criteria documented in LTR Section 1.3, using a novel application of the GOTHIC code. The staff found the TRACG M&E method acceptable based on it being consistent with the guidance in SRP Section 6.2.1.3 and found the method appropriately conservative for determining the M&E release, subject to the L&Cs noted in SE Section 7.0. The staff reviewed the PIRT developed for DBE evaluations with the BWRX-300 containment, the validation of the selected GOTHIC modeling to capture the relevant thermal-hydraulic phenomena, the sensitivity studies performed to assess the modeling choices in the LTR, and the uncertainty quantification. As a result of this review, the NRC staff found the proposed BWRX-300 analytical approach and GOTHIC modeling described in the LTR to be acceptable for their intended purpose, with the appropriate conservative biases and modeling inputs to address the model uncertainties.

Based on the above discussion, the NRC staff concludes that the analysis method presented in the LTR is acceptable for reference in future licensing activities to demonstrate that the BWRX-300 containment design can meet the acceptance criteria listed in LTR NEDC-33911P-A, Revision 3. This method is acceptable for use, consistent with the conditions and limitations noted in SE Section 7.0, in containment analysis of AOOs, station blackout, ATWS, LBLOCAs, and SBLOCAs.

As previously discussed in Section 1 of the SE, the NRC staff will evaluate the regulatory compliance of the final design of the containment and the final CEM for the BWRX-300 SMR during future licensing activities, in accordance with 10 CFR Part 50 or 10 CFR Part 52, as applicable.