ENCLOSURE 2

M210099

Responses to Requests for Additional Information eRAIs 9854, 9856, and 9862 and Supplemental Response to eRAI 9817

Licensing Topical Report
NEDC-33922P, Revision 0,
BWRX-300 Containment Evaluation Method

Non-Proprietary Information

**IMPORTANT NOTICE**

This is a non-proprietary version of Enclosure 1, from which the proprietary information has been removed. Portions of the enclosure that have been removed are indicated by an open and closed bracket as shown here [[ ]].

**SRP-Review Section: 06.02.01 - Containment Functional Design Application Section:**

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**06.02.01-02 (eRAI 9862) [Audit Issue 2]**
**Date of eRAI Issue: 08/05/2021**

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**Requirement**

General Design Criterion 50 – *Containment design basis*. Requires 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).

General Design Criterion 38 -- *Containment heat removal*. A system to remove heat from the reactor containment shall be provided. The system safety function shall be to reduce rapidly, consistent with the functioning of other associated systems, the containment pressure and temperature following any loss-of-coolant accident and maintain them at acceptably low levels.

General Design Criterion 16 -- *Containment design*. Reactor containment and associated systems shall be provided to establish an essentially leak-tight barrier against the uncontrolled release of radioactivity to the environment and to assure that the containment design conditions important to safety are not exceeded for as long as postulated accident conditions require.

**Issue**

GEH LTR NEDC-33922P, Revision 0, BWRX-300 Containment Evaluation (CE) Method, presents TRACG and GOTHIC results for the conservative case of the small steam pipe break design basis event (DBE), as the limiting small break LOCA (SBLOCA) for BWRX-300. According to the LTR, this break in a pipe of [[ ]] attached to the RPV dome bounds all small steam pipe breaks that remain unisolated. Figure 5-18 through Figure 5-22 show various in-vessel TRACG results for the post-accident 72 hours, including power, RPV pressure, RPV level, PCT, break flow rate and enthalpy. The LTR also includes the necessary GOTHIC results for the same DBE in Figure 6-31 through 6-40 to demonstrate the resulting containment thermal hydraulic response, steam stratification, radiolytic gas accumulation inside the containment/dome, and PCCS/reactor cavity pool environment characteristics.

However, GEH LTR NEDC-33922P presents only TRACG results for the conservative case of the small liquid pipe break DBE. Figure 5-23 through Figure 5-27 show similar in-vessel TRACG calculation results for the small liquid pipe break DBE, as the aforementioned Figure 5-18 through Figure 5-22 for the small steam pipe break DBE. However, no related GOTHIC results are presented for the small liquid pipe break DBE in the LTR similar to Figure 6-31 through 6-40 for the small steam pipe break DBE. Therefore, the results presented in the LTR for the small liquid pipe break DBE are an incomplete set of the similar results presented in the LTR for the small steam pipe break DBE. In the absence of this information, the staff is unable to make a reasonable assurance finding for the bounding nature of the conservative case for the small steam pipe break.

**Request**

The applicant is requested to provide the GOTHIC results for the BWRX-300 containment for the conservative case of the small liquid pipe break similar to those provided for the small steam pipe break, and justify how the limiting small steam pipe break with the conservative case assumptions also bounds the most limiting small liquid pipe break.

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**GEH Response to NRC Question 06.02.01-02**

The base case results in the licensing topical report (LTR) show the steam pipe breaks are bounding for containment response. However, the conservative case multipliers affect the small liquid pipe breaks [[ ]], and as a result, conservative case results for the liquid and steam pipe break cases are similar to each other. Therefore, the small liquid break conservative cases will also be analyzed explicitly in the evaluation method. The results for these conservative case containment responses to the small liquid pipe break are presented in Figures 9862-1 through 9862-3. The statement that small steam breaks are the more limiting of the steam and liquid small pipe breaks will be removed from Section 6.10.2 of the LTR and the figures shown below will be added. All discussions in Section 6.10.2 are equally applicable to the small liquid breaks.

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**Figure 9862-1. Containment Pressure Following a Small Liquid Pipe Break**

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**Figure 9862-2. PCCS Exit and Reactor Cavity Pool Temperatures Following a Small Liquid Pipe Break, Conservative Case**

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**Figure 9862-3. Containment Temperatures Following a Small Liquid Pipe Break**

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**Proposed Changes to NEDC-33922P Revision 0**

The title of Section 6.10.2 will be changed to:

Containment Response to Small ~~Steam~~ Pipe Breaks, Conservative Cases

The first paragraph of Section 6.10.2 will be changed as follows:

The containment response predicted by using the conservative case assumptions for the small ~~steam~~ pipe breaks ~~which are the more limiting of the steam and liquid small pipe breaks,~~ is shown ~~on~~ in Figure 6-31 through Figure 6-34 for small steam pipe breaks and in Figure 6-39 through 6-41 for small liquid pipe breaks.

One paragraph will be added to Section 6.10.2 to discuss the containment response to small liquid pipe breaks. This paragraph will be provided to the NRC staff in a future submittal.

The following figures will be added to the end of Section 6.10.2:

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**Figure 6-39. Containment Pressure Following a Small Liquid Pipe Break**

[[

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**Figure 6-40. PCCS Exit and Reactor Cavity Pool Temperatures Following a Small Liquid Pipe Break, Conservative Case**

[[

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**Figure 6-41. Containment Temperatures Following a Small Liquid Pipe Break**

**06.02.01-03 (eRAI 9862) [Audit Issue 5]**
**Date of eRAI Issue: 08/05/2021**

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**Requirement**

General Design Criterion 50 – *Containment design basis*. Requires 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).

General Design Criterion 38 -- *Containment heat removal*. A system to remove heat from the reactor containment shall be provided. The system safety function shall be to reduce rapidly, consistent with the functioning of other associated systems, the containment pressure and temperature following any loss-of-coolant accident and maintain them at acceptably low levels.

General Design Criterion 16 -- *Containment design*. Reactor containment and associated systems shall be provided to establish an essentially leak-tight barrier against the uncontrolled release of radioactivity to the environment and to assure that the containment design conditions important to safety are not exceeded for as long as postulated accident conditions require.

**Issue**

Figure 6-11 in GEH LTR NEDC-33922P, Revision 0, BWRX-300 Containment Evaluation Method, presents the vertical and horizontal cross-sectional views of various 3-D GOTHIC grids used in the BWRX-300 containment nodalization studies. The LTR Figure 6-11 [[

]]. The LTR Figure 6-11 [[

]].

As recognized in LTR Section 6.6.1, [[

]]. The staff is concerned that the increased velocity and higher steam concentration near [[ ]] that increase the condensation heat transfer, would enhance the heat removal to the [[ ]]. While this has the desired effect of maximizing the shell temperature, it would have an adverse impact on the peak containment pressure. Therefore, the staff considers the current break location to be a non-conservative assumption embedded in the BWRX-300 GOTHIC model for the purpose of predicting the peak containment pressure.

The staff also needs to evaluate that, as based on the COPAIN test data presented in the LTR, the convection and condensation heat transfer coefficients [[

]]. So, GOTHIC's qualifications need to be justified for using the flow-direction/convection-mode dependent conservatisms used in the BWRX-300 CE methodology.

**Request**

1. The applicant is requested to provide the results for a break location sensitivity study for the BWRX-300 containment main steam pipe break (LBLOCA) using the conservative GOTHIC containment model to justify that the chosen break location is bounding for all break locations, or identify the most limiting break locations with respect to peak containment pressure and maximum wall temperature. Please perform the sensitivity study for the default [[ ]] for 24 hours, along the radial, axial, and azimuthal directions, or justify why some or all of the sensitivity cases are not needed.
2. The applicant is also requested to perform a 180 degree break flow orientation sensitivity study for the limiting location, i.e., the break flow coming out of the pipe upward, horizontal toward the containment wall, and downward directions, for the most limiting break location, as identified in Part 1, for the default [[ ]] conservative cases of the large steam break LBLOCA as well as the small steam break SBLOCA. The requested break orientation sensitivity study results for 24 hours for LBLOCA and 72 hours for SBLOCA, along with the information requested in Part 3, will help the staff make a reasonable assurance finding regarding the conservative biases used in modeling convection and condensation to cover the uncertainties in the COPAIN test data, as documented in the LTR. The staff needs to evaluate the [[ ]] biases used for the convection and condensation heat transfer correlations.
3. [[ ]], the applicant is requested to provide justification for GOTHIC's qualification to predict the flow direction in the near wall region in a subdivided volume, or provide evidence that in the limiting cases GOTHIC's prediction of flow direction is in the conservative direction in the near wall region of the subdivided containment. Otherwise, present the limiting LBLOCA and SBLOCA results using [[ ]].

Please update the LTR if the responses to the above questions result in a change to the methodology as described in the LTR.

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**GEH Response to NRC Question 06.02.01-03**

The responses to each item are presented below:

1. Break location sensitivity for a large steam pipe break was evaluated by changing the break location as listed in Table 9862-1. The case, [[ ]] is the same case that is presented in Section 6.10.1 of the licensing topical report (LTR) for a large steam pipe break, conservative case.

   The break locations are shown in Figure 9862-4. The large break cases are run up to four (4) hours. Because the break is isolated in [[ ]]. Because the differences in the results are due to the differences in break location and the break flow was terminated [[ ]], the trends will continue to converge over time. Therefore, it was determined that no further relevant information would be gained by extending the analyses to 24 hours.

   **Table 9862-1. Break Location Cases for a Large Steam Pipe Break**

   | Case Identifier | Description |
   |---|---|
   | [[ ]] | Break is at [[ ]], placed near a wall and directed towards the wall. This is the same break location and orientation as in NEDC-33922P Revision 0. |
   | [[ ]] | Break is at [[ ]], placed near the reactor pressure vessel (RPV), directed towards the wall. |
   | [[ ]] | Break is at [[ ]], placed near the RPV, directed towards the wall. |

   The containment pressure and temperature are shown in Figures 9862-5 and 9862-6. The containment pressure shown in Figure 9862-5 [[ ]]. The reason for this can be seen in Figure 9862-6. [[ ]]

   A sensitivity case in the azimuthal direction was not performed because the containment is circular and [[ ]].

2. The sensitivity to the break orientation for the most limiting break location case above [[ ]] was evaluated for a large steam pipe break, as listed in Table 9862-2. The

results are plotted in Figure 9862-7. [[

]] Therefore, a small break orientation sensitivity study was not performed.

However, a break location sensitivity case was performed for a small steam pipe break as listed in Table 9862-3. The results are compared in Figure 9862-8. [[

]] The peak containment pressure resulting from small breaks is far below the peak containment pressure resulting from large breaks. The only concern with respect to the containment response for small breaks is the elevated pressure over the long-term (i.e., one day or longer). There is also a large margin in the long-term results as discussed below.

The containment back pressure is not credited in calculating the small break mass and energy release so as to bound potential scenarios where the containment may leak, normal containment cooling may be running during the event, or the break may be outside the containment. Therefore, it may not be justifiable to credit containment back pressure in all cases for the core cooling response. [[

]] However, the results presented in LTR Figure 6-38 show that [[

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**Table 9862-2. Break Flow Orientation Sensitivity Cases for a Large Steam Pipe Break**

| Case Identifier | Description |
|---|---|
| [[ ]] | Break is at [[ ]], placed near the RPV, oriented towards the shell. |
| [[ ]] | Break is at [[ ]], placed near the RPV, oriented upwards. |
| [[ ]] | Break is at [[ ]], placed near the RPV, oriented downwards. |

**Table 9862-3. Break Flow Location Sensitivity Cases for a Small Steam Pipe Break**

| Case Identifier | Description |
|---|---|
| [[ ]] | Break is at [[ ]], placed near the shell, oriented towards the shell. This is the same break location and orientation as in LTR Section 6.10.2. |
| [[ ]] | Break is at [[ ]], placed near the RPV, oriented upwards. |

3. The GOTHIC validation includes several comparisons with experimental test data for situations that involve multidimensional flows where buoyancy forces are significant, as described below. Examples 1 and 5 include direct comparison of the measured and calculated velocity. For the other tests, reasonable agreement with the test velocities can be inferred from the comparisons of the measured local temperature with the GOTHIC calculated temperatures. These tests are documented in Reference R9862-1. Taken together they show that GOTHIC has the fundamental capability to calculate the velocity field in a vessel due to local jets and buoyancy.

    1. GOTHIC provides good agreement for the steady velocity profile in a thermally driven cavity as presented in Section 5.5.1 of Reference R9862-1. The test from Reference R9862-2 is for two-dimensional flow in an air-filled cavity (2.5m high x 0.5 m wide). The top and bottom of the cavity were insulated, and the sides were maintained at 72.9°C and 27.1°C. Based on the peak velocity and the wall spacing, the Reynolds (Re) number for this test is ~10,000 and the Grashof (Gr) number is ~$8x10^8$. The velocity comparison is shown in Figure 5.14 in Reference R9862-1.

    2. Comparisons to a two-dimensional natural convection from a horizontal cylinder in a rectangular cavity is presented in Section 5.6 of Reference R9862-1. This case is two-dimensional natural convection from a heated horizontal cylinder in a rectangular cavity with cooled side walls, an insulated bottom and convective heat loss from the top. The GOTHIC predicted Nusselt number for the overall heat transfer from the cylinder agrees with the measured data to within 10% over a Rayleigh (Ra) range of 1300 to 3400. The discrepancy is within the uncertainty of the experiment.

    3. GOTHIC was used to model a mixed convection test in three different test facilities that made up International Standard Problem 47 in References R9862-3 and R9862-4 as presented in Section 15 of Reference R9862-1. The TOSQAN test (Volume (V)=7 $m^3$, Height (H)=5 m) and the MISTRA test (V=100 $m^3$, H=7.4 m) were axisymmetric with an upward vertical steam or steam and helium jet at the vessel center and cooled vessel wall. The THAI test (V=60 $m^3$, H=9.2 m) was three-dimensional with upward vertical and horizontal steam and helium injection and a cooled vessel wall. Local Re and Gr numbers varied widely throughout the vessel for these tests, covering forced, mixed and free

convection regimes. GOTHIC generally compares well with the local temperature and concentration measurements indicating that the velocity field is reasonably predicted.

4. GOTHIC was used to simulate the HEDL experiments for hydrogen mixing in a scaled ice condenser containment as presented in Section 9 of Reference R9862-1. The tests included vertical upward and horizontal hydrogen jets that evolved to buoyant plumes, with and without mixing fans. Generally good agreement was obtained with the measured local hydrogen concentration indicating that the velocity field was reasonably well predicted.
5. GOTHIC benchmarking to the velocity profiles for a two-dimensional turbulent jet (slot jet) is presented in Section 5.9 of Reference R9862-1. The GOTHIC comparison with the centerline velocity versus distance from the slot for a selected test is shown in Figure 5.33 of Reference R9862-1 and shows good agreement.

The test cases above demonstrate GOTHIC's capability to calculate the velocity field resulting from jets and buoyancy.

For the small steam break case, [[ ]], in Figure 9862-5 the velocities at each PCCS location along the entire height of the containment are shown in Figures 9862-9a and 9862-9b at approximately the time of the peak pressure [[ ]], and at one (1) day. [[ ]]

[[ ]] Relaminarization simply does not occur if the flow direction is counter to the direction of the buoyancy forces as discussed in Section 6.8.2 of the LTR. The containment evaluation method presented in the LTR also superimposes each source of conservatism, compounding the effects of the biases. Furthermore, the assumption that the break mass flow rate is not affected by the containment back pressure alone adds more than [[ ]]. When added together, any variations resulting from [[ ]] are much smaller than the overall conservatism in the method for containment response to small breaks.

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**Figure 9862-4. Break Location Sensitivity Study, Break Locations for Steam Pipe Breaks.**

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**Figure 9862-5. Containment Pressure Sensitivity to Break Location for Large Steam Pipe Break. (See Figure 9862-4 for Legend).**

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**Figure 9862-6. Containment Temperature Sensitivity to Break Location for Large Steam Pipe Break. (See Figure 9862-4 for Legend).**

[[

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**Figure 9862-7. Containment Pressure Sensitivity to Break Orientation for Large Steam Pipe Break. (See Figure 9862-4 for Legend).**

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**Figure 9862-8. Containment Pressure Sensitivity to Break Location for a Small Steam Pipe Break. (See Figure 9862-4 for Legend).**

[[

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**Figure 9862-9a. Velocities at the PCCS Locations from Containment Bottom to Top at 7200 seconds. Maximum Velocity:** [[ ]]**. Small Steam Pipe Break** [[ ]] **in Figure 9862-8.**

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**Figure 9862-9b. Velocities at the PCCS Locations from Containment Bottom to Top at 86400 seconds. Maximum Velocity:** [[ ]]**. Small Steam Pipe Break Case,** [[ ]] **in Figure 9862-8.**

## References

R9862-1. GOTHIC Thermal Hydraulic Analysis Package, Version 8.3(QA), Qualification Report, EPRI, Palo Alto, CA: 2018.

R9862-2. R Cheesewright, KJ King, and S Ziai. Experimental Data for the Validation of Computer Codes for the Prediction of Two-Dimensional Buoyant Cavity Flows, in Significant Questions in Buoyancy Affected Enclosure or Cavity Flows. Technical report, ASME Winter Annual Meeting, Anaheim, CA, December 1986. HTD Vol. 60.

R9862-3. P Cornet, J Malet, E Porcheron, J Vendel, E Studer, and M Caron-Charles. ISP-47 Specification of International standard problem on containment thermal hydraulics, Step 1: TOSQAN-MISTRA. Technical report, Institut de Radioprotection et de Surete Nucleaire, Saclay, France, July 2002. Revision 1, DPEA/SERAC/LPMAC/02-44.

R9862-4. K Fischer et al. International Standard Problem ISP-47 on Containment Thermal-Hydraulics, Step 2: ThAI, Comparison Report of Blind Phases I-IV. Technical report, Becker Technologies GmbH, Eschborn, January 2005. BF-R 70031-2.

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**Proposed Changes to NEDC-33922P Revision 0**

The pressures for the conservative case in Figure 6-26, and all cases in Figures 6-28 and 6-31 will be based on the break location near the RPV and at [[ ]]. This updated information will be provided to the NRC staff in a future submittal.

A statement will be added to Sections 6.10.1 and 6.10.2 to explain that break location near the RPV is used to maximize the containment pressure, and break location near the shell is used to maximize the shell temperature. This text will be provided to the NRC staff in a future submittal.

In Section 6.11, the following bullets will be added under the "conservative containment cases":

- [[ ]]
- [[ ]]

**06.02.01-04 (eRAI 9862) [Audit Issue 36]**
**Date of eRAI Issue: 08/05/2021**

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**Requirement**

General Design Criterion 38 -- *Containment heat removal*. A system to remove heat from the reactor containment shall be provided. The system safety function shall be to reduce rapidly, consistent with the functioning of other associated systems, the containment pressure and temperature following any loss-of-coolant accident and maintain them at acceptably low levels.

General Design Criterion 16 -- *Containment design*. Reactor containment and associated systems shall be provided to establish an essentially leak-tight barrier against the uncontrolled release of radioactivity to the environment and to assure that the containment design conditions important to safety are not exceeded for as long as postulated accident conditions require.

**Issue**

Section 2.0 of the GEH LTR NEDC-33922P, Revision 0, BWRX-300 Containment Evaluation (CE) Method, states that [[

]]. The GOTHIC models submitted with this application, the sensitivity studies, results, and descriptions documented in the LTR, the correlations the BWRX-300 CE methodology relies on, as well as the staff review and confirmatory analyses are all based on [[

]]. The study presented in the LTR for the effect of nodalization on a single PCCS unit's performance is also based on [[

]].

However, the LTR also mentions a [[ ]] design option, and LTR Section 6.1 states [[

]]) to establish the applicability of the BWRX-300 PCCS phenomenology independent of the PCCS design configuration. The staff is concerned that without a supplementary review of any new PCCS design configuration, the associated model for secondary side heat transfer to the reactor cavity pool, and supporting containment/PCCS results, the staff may not be able make a reasonable assurance finding about the applicability of the BWRX-300 CE methodology to the new PCCS design.

**Request**

The applicant is requested to provide justification to extend the BWRX-300 CE methodology that was reviewed for the [[

]] without a supplementary NRC review. Otherwise, please remove all LTR references to the [[

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**GEH Response to NRC Question 06.02.01-04**

The water jacket configuration had been a potential option earlier in the development of the BWRX-300 containment design. However, the water jacket design presented challenges to the [[ ]] and has been abandoned. The water jacket is no longer a design option.

Currently, two variations of the pipe configuration are being considered. The GOTHIC methodology is capable of analyzing either configuration. In the concentric pipe configuration shown in Figure 2-4 of NEDC-33922P, the cold pipe is enclosed inside the hot channel. This configuration requires that each unit has a different penetration. [[

]] Heat transfer occurs on the vertical PCCS pipes, the same geometry as in the concentric pipe configuration. [[ ]]

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**Proposed Changes to NEDC-33922P Revision 0**

None.

**06.02.01-05 (eRAI 9862) [Audit Issue 37]**
**Date of eRAI Issue: 08/05/2021**

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**Requirement**

General Design Criterion 38 -- *Containment heat removal*. A system to remove heat from the reactor containment shall be provided. The system safety function shall be to reduce rapidly, consistent with the functioning of other associated systems, the containment pressure and temperature following any loss-of-coolant accident and maintain them at acceptably low levels.

General Design Criterion 16 -- *Containment design*. Reactor containment and associated systems shall be provided to establish an essentially leak-tight barrier against the uncontrolled release of radioactivity to the environment and to assure that the containment design conditions important to safety are not exceeded for as long as postulated accident conditions require.

**Issue**

Table 6-2 of the GEH LTR NEDC-33922P, Revision 0, BWRX-300 Containment Evaluation (CE) Method, presents the Phenomena Identification and Ranking Table for BWRX-300 containment. The table recognizes [[

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These phenomena are pertinent to the first 24 hours of LBLOCA and first 72 hours of SBLOCA. However, the staff found a lack of information in the LTR about the safety-related assumptions and modeling details about [[ ]]. Therefore, the applicant is requested to address the following questions about [[ ]] modeling.

**Request**

1. Section 6.10.2 of the GEH LTR NEDC-33922P, Revision 0, BWRX-300 Containment Evaluation (CE) Method, states “the calculations conservatively assume no heat loss from the reactor cavity pool to the surroundings.” In addition, the LTR on Page 59 refers to [[ ]]. However, LTR Figure 6-1 shows [[ ]] So, the LTR does not recognize [[ ]] The applicant is requested to provide a clarification regarding [[ ]] and document the reactor cavity pool’s exposure to the ambient atmosphere as a part of the CE methodology described in the LTR.
2. Please provide and justify the assumptions made in the BWRX-300 CE methodology about modeling the above-mentioned [[ ]] PIRT phenomena as identified in the

LTR Table 6-2, and describe how they were addressed in the model. A review of the GOTHIC model submitted with the application shows [[

]] The LTR does not include these CE methodology details and justifications, lacks information on the ambient temperature and initial humidity conditions, and is unclear as to whether [[ ]] is a part of the model. The applicant is requested to document the modeling details as part of the BWRX- 300 CE methodology description.

3. Even though the LTR does not explicitly state as such, the staff infers that the BWRX-300 CE methodology review scope is limited to the first 72 hours after the initiation of the postulated design basis events (DBEs). [[
]]. Therefore, the staff requests the applicant to clarify the staff's understanding regarding the LTR review scope being limited to the first 72 hours of the postulated DBEs. Otherwise, provide additional information to justify that the BWRX-300 containment design would remain safe with sufficient cooling mechanisms beyond 72 hours.

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**GEH Response to NRC Question 06.02.01-05**

1. Heat loss from the pool to the surroundings through the walls are not included. However, there is some heat loss by evaporation at the surface of the pool.

   Using the small steam pipe break results, the following data are obtained from the GOTHIC output, where Volume 4 in the model is comprised of the reactor cavity pool and the reactor building airspace volume above the refuel floor. This volume is initially filled with water in the pool and air at atmospheric pressure in the airspace.

| [[ | | |
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The pool mass, $M_{pool}$, is calculated from the GOTHIC results as,

$$M_{pool} = V\left(1-\alpha\right)\rho_{\ell}$$

where [[ ]] is the volume of the pool and the airspace in the refueling floor, $\alpha$ is the volume fraction of steam/air and $\rho_{\ell}$ is the density of liquid.

At the initial and final conditions, the mass of the pool is calculated as

[[

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The amount of energy lost to evaporation is approximately [[ ]] of the energy deposited in the pool. The majority of the energy lost to evaporation is not released to the environment but remains in the refueling floor atmosphere. Although the energy lost to evaporation is a small fraction, the statement in the licensing topical report (LTR) will be revised by stating that the energy loss from the pool due to surface evaporation is accounted for.

2. The passive containment cooling system (PCCS) intake location in the pool is near the bottom of the pool. The PCCS return pipe discharge elevation is at least [[ ]] the PCCS intake elevation. Hot water rising from the discharge of the PCCS return pipe may cause some stratification in the pool. If the stratification is significant, the water temperature at the intake location of the PCCS would be lower than the bulk pool temperature. It is not plausible that stratification would occur in such a manner that hot water would collect at the bottom of the pool. Ignoring this temperature difference and using a lumped parameter model for the pool is more conservative than resolving the temperature gradients in the pool.

   The initial humidity in the pool area was assumed to be [[ ]] in the demonstration calculations. This has little effect on the calculations. The LTR will include the following requirements for the application method:

   - The initial airspace temperature above the reactor cavity pool is assumed to be the same as the pool water temperature.
   - The initial relative humidity of the airspace is assumed to be 100%.
   - The reactor cavity pool is modeled as a lumped parameter volume. The air space is connected to a constant pressure boundary condition such that the airspace pressure is nearly constant at atmospheric pressure.

3. The applicability of the LTR is not based on a time limit, but rather on the phenomena that are modeled and the applicability of the biases. The heat removal capacity of the PCCS will

decrease as the pool heats up, but the model does not have a particular limitation on the heat transfer modeling up to the point where [[ ]]. Also, the capability of the model has not been demonstrated for degraded event analyses where the steam released from the reactor pressure vessel (RPV) may be significantly superheated or may contain significant amounts of hydrogen. Significant superheating or significant hydrogen generation does not occur within 72 hours during a design basis accident.

Although GOTHIC is fundamentally capable of modeling the phenomena if significant hydrogen generation and/or significant superheat occurs, the LTR does not attempt to quantify the additional uncertainties, if any, resulting in such progression of an accident, and GEH does not seek approval for this quantification of additional uncertainties in this LTR.

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**Proposed Changes to NEDC-33922P Revision 0**

The last sentence in the 2nd paragraph of Section 6.10.2 will be revised as follows:

Note that the calculations conservatively assume no heat loss from the reactor cavity pool to the surroundings through the walls. However, heat loss due to surface evaporation is accounted for.

The following bullets will be added to Section 6.11:

- The initial airspace temperature above the reactor cavity pool is assumed to be the same as the pool water temperature.
- The initial relative humidity of the airspace is assumed to be 100%.
- The reactor cavity pool is modeled as a lumped parameter volume. The air space is connected to a constant pressure boundary condition such that the airspace pressure is nearly constant at atmospheric pressure.

**06.02.01-09 (eRAI 9856) [Audit Issue 21]**
**Date of eRAI Issue: 08/05/2021**

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**Requirement**

General Design Criterion 50 – Containment design basis, requires 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).

**Issue**

In order to determine the conservative mass and energy discharge to the containment, a computer code and the associated evaluation model needs to have the capability to model relevant physical phenomenon during a LOCA with a conservative treatment of uncertainties. Standard Review Plan (NUREG-0800) Section 6.2.1.3, "Mass and Energy Release Analysis for Postulated Loss-of-Coolant Accidents (LOCAs)," notes that "calculations of the mass and energy release rates for a LOCA should be performed in a manner that conservatively establishes the containment internal design pressure (i.e., maximizes the post-accident containment pressure and the containment sub-compartment response)."

In response to staff RAI #9817, GEH proposed a conceptual design change to limit the non-condensable gas volume concentration [[ ]] below a safe threshold value. Possible conceptual changes include [[ ]], the combustible gas recombining could generate heat upon the actuation of Isolation Condensers. The current SBLOCA and LBLOCA TRACG model including the Isolation Condenser model [[ ]].

**Request**

Therefore, the staff is requesting additional information regarding the potential heat addition [[ ]] and the potential impact on the ICs heat removal capacity when the non-condensable gas concentration in the lower drum is maintained up to the safe threshold value.

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**GEH Response to NRC Question 06.02.01-09**

The radiolytic gas control design is currently in process, and the amount of energy that will be added to the steam [[ ]], if any, has yet to be determined. Therefore, a bounding estimate is presented herein assuming all of the energy that can theoretically be released from the recombination of hydrogen and oxygen at the stoichiometric ratio will be added to the ICS for the cases presented in the GEH response to RAI 9817.

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The cases presented in the GEH response to RAI 9817 were re-run to account for the energy addition described above. The ratio of the energy release rate from recombination of radiolytic gases to the ICS heat removal rate is shown in Figure 9856-1 for the two cases presented in the GEH response to RAI 9817. [[

]] Therefore, the radiolytic gas fraction at the inlet of the isolation condenser is not diluted in this calculation and increases continuously. Despite this added conservatism, the energy addition [[ ]] is still a small fraction of the heat removal rate of the isolation condenser.

The ICS heat removal rates are compared with and without energy addition in Figure 9856-2. The heat removal rates are indistinguishable from each other, indicating that [[ ]] has practically no effect on the heat removal rate as discussed above.

[[

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**Figure 9856-1. Ratio of the Energy Release Rate from Recombination of Radiolytic Gases to the Heat Removal Rate of Isolation Condenser.**

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**Figure 9856-2. Isolation Condenser Heat Removal Rate With and Without Recombination Energy**

**Reference**

R2856-1. Kenneth K. Kuo, "Principles of Combustion," 2nd Edition, John Wiley & Sons, 2005.

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**Proposed Changes to NEDC-33922P Revision 0**

None.

**SRP-Review Section: 06.02.05 - Combustible Gas Control in Containment**

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**06.02.05-01 (eRAI 9854) [Audit Issue 14]**
**Date of eRAI Issue: 08/05/2021**

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**Requirement**

The NRC regulations in 10 CFR 50.44(c) set forth combustible gas control requirements for future water-cooled nuclear power reactor designs. In accordance with SRP Section 6.2.5, the NRC staff reviewed the BWRX-300 containment design for consistency with 10 CFR 50.44 (c).

**Issue**

To meet 10 CFR 50.44 (c), Section 6.10.3 of the LTR (NEDC-33922P, Revision 0) states that the calculated hydrogen and oxygen volume fractions are far below the "deflagration limits." In addition, the BWRX-300 is designed to have an inert containment such that the oxygen concentration must be limited. GEH indicated in electronic reading room (eRR) [[

]] to meet the definition of inerted atmosphere in 10 CFR 50.44(c).
[[ ]]

**Request**

Since these quantitative limits are necessary for the combustible gas control in containment and the audit files in the eRR are not on docket nor are referenced in the LTR, the applicant is requested to confirm or specify the quantitative hydrogen and oxygen concentration limits being used to ensure the compliance of 10 CFR 50.44(c) for combustible gas control in containment on docket.

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**GEH Response to NRC Question 06.02.05-01**

The licensing topical report (LTR) does not try to set and does not assume combustible gas fraction limits or deflagration limits. Note also that 10 CFR 50.44(c) only requires that the initial oxygen volume fraction be kept below 4% for an inerted containment. There is no regulatory requirement for a hydrogen volume fraction.

The statement in Section 6.10.3 of the LTR with respect to the radiolytic gas fractions being well below deflagration limits refers to the order of magnitude of the volume fractions that may cause deflagration rather than the precise values.

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.

Hydrogen deflagration limits are discussed in References R9854-1, R9854-2 and R9854-3.

Reference R9854-3 states that the flammability limit for hydrogen in dry air is 4%. Combustion is inhibited above 75% hydrogen concentration. The minimum oxygen required for combustion to take place is 5%. The autoignition temperature, whether it is the gas temperature or the surface temperature, is in the range of 580 – 800°C. Figure 1.2.3-1 of Reference R9854-2 shows the flammability limits for upward propagation and downward propagation at two different pressures and various temperatures. As shown in this figure, the downward propagation limit for the flammability of hydrogen in dry air is at least 8%, which would be representative of hydrogen collecting at the top of the containment if the containment were filled with dry air.

The hydrogen burn in the Three Mile Island (TMI) containment was estimated to result from a hydrogen accumulation of 7.3 – 7.9% in the containment (Appendix S of Reference R9854-1). The TMI containment is not inerted. Table S.3-2 of Reference R9854-1 shows that the minimum hydrogen concentration required for combustion increases significantly when the steam fraction in dry air increases. When the steam fraction is 40%, the required hydrogen fraction in the mixture increases to as much as 15 to 40%. Combustion is inhibited at even higher steam concentrations (Table S.3-1 of Reference R9854-1). At high steam concentrations, combustion is incomplete even if it could occur.

The LTR asserts that the radiolytic gas volume fractions in the BWRX-300 containment are far below any of the values quoted above but are not compared to a specific limit.

**References**

R9854-1. EPRI Final Report, 1025295, "Severe Accident Management Guidance Technical Basis Report, Volume 2: The Physics of Accident Progression," October 2012.

R9854-2. NEA/CSNI/R(2014)8, "Status Report on Hydrogen Management and Related Computer Codes," June 2014.

R9854-3. NEA/CSNI/R(2000)10, "Carbon Monoxide – Hydrogen Combustion Characteristics in Severe Accident Containment Conditions," March 2000.

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**Proposed Changes to NEDC-33922P Revision 0**

None

**SRP-Review Section: 06.02.01 - Containment Functional Design Application Section:**

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**06.02.01-01 (eRAI 9817)**
**Date of eRAI Issue: 04/08/2021**

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**Requirement**

General Design Criterion 50 – Containment design basis, requires 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).

**Issue**

In order to determine the conservative mass and energy discharge to the containment, a computer code and the associated evaluation model needs to have the capability to model relevant physical phenomenon during a LOCA with a conservative treatment of uncertainties. Standard Review Plan (NUREG-0800) Section 6.2.1.3, "Mass and Energy Release Analysis for Postulated Loss-of-Coolant Accidents (LOCAs)," notes that "calculations of the mass and energy release rates for a LOCA should be performed in a manner that conservatively establishes the containment internal design pressure (i.e., maximizes the post-accident containment pressure and the containment subcompartment response)."

GEH states in Section 5.2.4 of licensing topical report "BWRX-300 Containment Evaluation Method (NEDC-33922P, Revision 0)," that the only non-condensable gases that may migrate into the isolation condenser system (ICS) tube bundles are the radiolysis products following a design basis LOCA. This is based on the design not experiencing any significant fuel cladding oxidation during a LOCA.

However, based on the BWRX-300 ICS design, [[

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**Request**

Therefore, the staff is requesting additional information regarding [[ ]] the associated modeling uncertainties, and the subsequent consequences for both large break and small break LOCA limiting cases.

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**GEH Supplemental Response to NRC Question 06.02.01-01**

The isolation condenser heat removal rate is calculated using a stand-alone isolation condenser model with the boundary conditions obtained from the conservative small break case shown in Figures 5-18 through 5-22 of the licensing topical report (LTR). The stand-alone isolation condenser model used for this purpose is shown in Figure 9817-4 below to simulate a recombiner to control the volumetric fraction of radiolytic gases [[

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The boundary conditions used in the stand-alone isolation condenser are shown in Figure 9817-5. The reactor pressure vessel (RPV) dome pressure is obtained from the conservative small break results. The pressure of the chimney boundary condition is set sufficiently higher than the pressure at the dome boundary condition such that liquid does not clear from the condensate return line trapping all radiolytic gases in the isolation condenser. The radiolytic gas volume fraction at the boundary conditions representing the RPV is also obtained from the conservative small steam break case.

It is assumed that a recombiner maintains [[ ]]. The heat removal rates calculated for the two cases are compared to the clean isolation condenser case in which the radiolytic gas volume fraction is set to zero at the boundary conditions representing the RPV. The comparisons are plotted in Figure 9817-6. The heat removal rates are indistinguishable from each other, indicating practically no effect of the radiolytic gas build up on the isolation condenser heat removal rate.

The reason that there is practically no effect of radiolytic gases on the heat removal rate can be deduced from Figures 9817-7 and 9817-8. Although the radiolytic gas volume fraction is allowed to build up to [[

]].

The above results show that there is practically no effect of radiolytic gases on the isolation condenser performance as long as the radiolytic gas volume fraction is maintained below a sufficiently low value.

The results presented above are also applicable to the large break cases. The difference between the large and small break cases is the radiolytic gas volume fraction at the boundary condition of the stand-alone isolation condenser model. The radiolytic gas volume fraction at the boundary condition would be somewhat higher in the large break cases as calculated in this analysis. [[ ]] Because of this assumption, the method already has a substantial conservatism built in and, therefore, the results presented for a small break case bounds the large break cases. It should also be noted that some degradation in the isolation condenser heat removal capacity for large breaks does not have an adverse effect in the long term response. It has only a minor effect on the final value of the long-term RPV pressure.

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**Figure 9817-4. Stand-Alone Isolation Condenser Model Used to Demonstrate Isolation Condenser Performance with Radiolytic Gas Control**

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**Figure 9817-5. Boundary Conditions Obtained from NEDC-33922P Conservative Small Steam Break Case, and the Increased Chimney Pressure to Maintain a Liquid Plug in Condensate Return Pipe**

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**Figure 9817-6. Isolation Condenser Heat Removal Rate Comparisons**

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**Figure 9817-7. Radiolytic Gas Volume Fraction in the Isolation Condenser.** [[
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**Figure 9817-8. Radiolytic Gas Volume Fraction in the Isolation Condenser.** [[
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