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RS-18-022

10 CFR 50.90

February 14, 2018

U.S. Nuclear Regulatory Commission
ATTN: Document Control Desk
Washington, DC 20555-0001

Dresden Nuclear Power Station, Units 2 and 3
Renewed Facility Operating License Nos. DPR-19 and DPR-25
NRC Docket Nos. 50-237 and 50-249

Subject: Response to NRC Request for Additional Information for Request for License Amendment to Revise Technical Specification 5.5.12, "Primary Containment Leakage Rate Testing Program"

References: (1) Letter from P. R. Simpson (Exelon Generation Company, LLC (EGC)) to NRC, "Request for License Amendment to Revise Technical Specifications Section 5.5.12 for Permanent Extension of Type A and Type C Leak Rate Test Frequencies," dated May 3, 2017

(2) Email from R. S. Haskell II (NRC) to M. A. Mathews (EGC), "Dresden Nuclear Power Station, Units 2 and 3 - RAIs to Support Request to Revise TS 5.5.12, 'Primary Containment Leakage Rate Testing Program' (EPID: L-2017-LLR-0228)," dated December 22, 2017

In Reference 1, Exelon Generation Company, LLC submitted a request for amendments to Renewed Facility Operating License Nos. DPR-19 and DPR-25 for Dresden Nuclear Power Station (DNPS), Units 2 and 3. Specifically, EGC requested that the NRC completed its review and approval of the request to modify Technical Specification (TS) 5.5.12, "Primary Containment Leakage Rate Testing Program," to allow for the permanent extension of the Type A Integrated Leak Rate Testing (ILRT) and Type C Leak Rate Testing frequencies.

In Reference 2, the NRC determined that additional information is required to complete the evaluation of the Reference 1 request. EGC's response to Reference 2 is contained in the attachment to this letter.

EGC has reviewed the information supporting a finding of no significant hazards consideration, and the environmental consideration, that were previously provided to the NRC in Reference 1. The additional information provided in this submittal does not affect the bases for concluding that the proposed license amendment does not involve a significant hazards consideration. In addition, the information provided in this submittal does not affect the bases for concluding that neither an environmental impact statement nor an environmental assessment needs to be prepared in connection with the proposed amendment.

EGC is notifying the State of Illinois of this supplement to a previous application for a change to the TS by sending a copy of this letter and its attachment to the designated State Official in accordance with 10 CFR 50.91, "Notice for public comment; State consultation," Paragraph (b).

There are no regulatory commitments contained in this letter. Should you have any questions concerning this letter, please contact Mr. Mitchel A. Mathews at (630) 657-2819.

I declare under penalty of perjury that the foregoing is true and correct. Executed on the 14th day of February 2018.

Respectfully,

Patrick R. Simpson
Manager – Licensing

Attachment: Response to Request for Additional Information

cc: NRC Regional Administrator, Region III
NRC Senior Resident Inspector – Dresden Nuclear Power Station
Illinois Emergency Management Agency – Division of Nuclear Safety

ATTACHMENT

Response to Request for Additional Information

<u>REQUEST FOR ADDITIONAL INFORMATION</u>

<u>LICENSE AMENDMENT REQUEST – REGARDING EXTENSION OF THE 10 CFR 50 APPENDIX J CONTAINMENT TYPE A & TYPE C TEST INTERVALS DRESDEN NUCLEAR POWER STATION UNITS 2 & 3</u>

<u>DOCKET NOS. 50-237 & 50-249</u>

By letter dated May 3, 2017, (Agencywide Documents Access and Management System (ADAMS) Accession No. ML17123A104), Exelon Generation Company, LLC, (the licensee) submitted a license amendment request to revise Dresden Nuclear Power Station, Unit 2 and 3 (DNPS) Technical Specification 5.5.12. The proposed amendment would extend the existing Type A integrated leakage rate test program test interval from 10 years to 15 years, and to adopt an extension of the containment isolation valve leakage rate testing (i.e., Type C test) frequency from 60 months to 75 months. The additional information below is needed to support the NRC staff's continued technical review of the license amendment request (LAR).

<u>PRA-RAI-1</u> Clarification of Probabilistic Risk Assessment (PRA) Model Version

In Table 5.1-2, "Accident Class 7 Failure Frequencies and Population Doses (DNPS Base Case Level 2 Model)," of Attachment 3 of the LAR (page 5-8), there is a reference to a 2014 PRA model in the column titled "2014 PRA Release Frequency / Yr." However, the 2013A PRA model appears to be used for the PRA evaluation supporting the LAR.

The NRC staff requests the licensee either correct the model referenced or justify the applicability of the model, as referenced in the LAR.

<u>Exelon Generation Company, LLC (EGC) Response:</u>

Table 5.1-2 incorrectly references a 2014 PRA model in the Column 3 header. The correct header for Table 5.1-2 is "2013A PRA RELEASE FREQUENCY / YR." This is an editorial correction and does not change any results or conclusions.

***PRA-RAI-2** Technical Adequacy of the Internal Flooding PRA Model*

In Section 3.4.2.5, "Consistency with Applicable PRA Standards," of Attachment 1 of the LAR (page 21 of 78), includes two focused scope peer reviews of the Internal Flooding PRA (IFPRA) model, one dated March 2009 and the other dated May 2010. In Section A.2.4, of Attachment 3 (page A-5) only discussed the March 2009 peer review, while in Table A-1, "2010 Focused Area IF Peer Review Findings/Status and Impact to Application," of Attachment 3 (beginning on page A-9) appears to provide the Findings and & Observations (F&Os) and associated resolutions from the May 2010 peer review. Furthermore, Section 3.4.2.5 identifies both of these peer reviews as having been done using NRC Regulatory Guide (RG) 1.200, "An Approach for Determining the Technical Adequacy of Probabilistic Risk Assessment Results for Risk-Informed Activities," Revision 1 (ADAMS Accession No. ML070240001). Section 3.4.2.5 states that an independent PRA peer review of the internal events PRA (IEPRA) was conducted in 2016 using RG 1.200, Revision 2 (ADAMS Accession No. ML090410014), and that all supporting requirements (SRs) were included except those related to internal flooding (which were previously peer reviewed in 2009).

The NRC staff uses Revision 2 of RG 1.200, to assess the technical adequacy of the PRA used to support risk-informed LARs. Therefore, the NRC staff requests that the licensee:

- *a. Provide a gap assessment of the IFPRA model against RG 1.200, Revision 2 in regards to this LAR.*
- *b. Provide justification why the IFPRA F&Os from the 2009 peer review were not provided in the LAR, or provide these F&Os to the NRC staff for review.*

EGC Response:

PRA-RAI-2 requests two deliverables: a gap assessment of the IFPRA model against RG 1.200, Revision 2 with regards to the Dresden Nuclear Power Station (DNPS), Units 2 and 3, Integrated Leak Rate Testing ILRT Frequency Extension license amendment request (LAR) and, justification for not providing F&Os from the IFPRA 2009 peer review or the request for additional information (RAI) that these F&Os be provided. The latter request (i.e., IFPRA 2009 peer review F&Os) is addressed first, followed by the response for a gap assessment of the IFPRA model against RG 1.200, Revision 2.

IFPRA 2009 peer review F&Os

Internal Flood F&Os

As noted above, PRA-RAI-2 requested the following: "Provide justification why the IFPRA F&Os from the 2009 peer review were not provided in the LAR, or provide these F&Os to the NRC staff for review."

The 2009 IFPRA Findings were provided in Table A-1 "2010 FOCUSED AREA IF PEER REVIEW FINDINGS/STATUS AND IMPACT TO APPLICATION" beginning on page A-9 of the License Amendment Request, RS-17-060, dated May 3, 2017. The title of Table A-1 was incorrect and should have referred to the 2009 peer review (i.e., there was no 2010 DNPS peer

review). All Internal Flood peer review findings from the 2009 peer review are contained in Table A-1.

It is also noted that Section 3.4.2.5 of the ILRT submittal listed, in error, a 2010 peer review. This section should have only listed the March 2009 peer review. The title change in Table A-1 of Attachment 3, "2010 Focused Area IF Peer Review Findings/Status and Impact to Application", to "2009 Focused Area IF Peer Review Findings/Status and impact to Application" is an editorial change.

Gap Assessment of the IFPRA Model Against RG 1.200, Revision 2

The March 2009 peer review was performed using NRC Regulatory Guide (RG) 1.200, "An Approach for Determining the Technical Adequacy of Probabilistic Risk Assessment Results for Risk-Informed Activities," Revision 1 (ADAMS Accession No. ML070240001). The March 2009 review was also performed using American Society of Mechanical Engineers (ASME)/ANS RA-Sb-2005, Addenda B to ASME/ANS RA-S-2002, Standard for Probabilistic Risk Assessment for Nuclear Power Plant Applications, dated December 2005. NRC RG 1.200 Revision 2 endorses ASME/ANS RA-SA-2009, Standard for Level 1/Large Early Release Frequency Probabilistic Risk Assessment for Nuclear Power Plant Applications, dated March 2009. Therefore, the following gap assessment addresses changes to the ASME Standard (December 2005 to March 2009), as well as RG 1.200 guidance (Revision 2).

Table RAI.2-1 documents a gap assessment between the Internal Flood related ASME Standard Requirements (SRs) in RA-SA-2009 and those included in ASME/ANS RA-Sb-2005, Addendum B. It is noted that the ASME RA-SA-2009 changes included a renumbering of SRs. The renumbering is captured in the first column of Table RAI.2-1, which provides each RA-SA-2009 SR number along with the corresponding RA-Sb-2005 SR number.

Specific to the treatment of uncertainties - the SRs from ASME RA-Sb-2005, Addendum B SR IF-F3 require the documenting of the sources of model uncertainty and related assumptions (as identified in QU-E1 and QU-E2) associated with the internal flooding analysis. In Table RAI.2-1, RA-SA-2009 SRs were added under each category relevant to an IF-F3 SR. In the Internal Flood evaluation, uncertainty and related assumptions highlighted by these SRs are addressed by the parametric uncertainty evaluation completed for the full PRA model. Assumptions are also identified in Section 2 of the DNPS Internal Flood Evaluation (DR PSA-013). Thus, these SRs are considered met and they have no impact on the DNPS ILRT submittal. See Table RAI.2-1 and notes following Table RAI.2-1 for additional detail.

RG 1.200 Revision 2 Review

Tables A-1, A-2 and A-3 of RG 1.200 Revision 2 identify the NRC issues, positions, and resolutions that were first assessed in relationship to the DNPS Internal Flood PRA modeling.

For the gap analysis, Table A-1 and Table A-2 issues, positions and resolutions were addressed first. In most cases, these issues and resolutions applied to the overall FPIE PRA model and were evaluated in the 2013A FPIE Self-Assessment as well as the 2016 FPIE Peer Review. A review of Tables A-1 and A-2 did not find any issues that would cause Internal Flooding requirements to be considered not met. An example of a review is provided below.

> Section 1-3: Clarification: Use of the word "significant" should match definitions provided in Section 2.2. Resolution: "The difference is not significant if the modeled accident sequences accounting of at least 95% of CDF/LERF for the hazard group…."

The internal flooding modeling is part of the overall FPIE PRA model. Therefore, SRs dealing with identification of significant cutsets, sequences, Human Reliability Analysis (HRA) contributors, etc. are evaluated with internal flooding as an input. The applicable SRs have been evaluated and found to meet Capability Category (CAT) I or above in the Self-Assessment performed on the 2013A FPIE PRA model as well as in the subsequent 2016 FPIE peer review.

Table A-3 issues, positions and resolutions align with ASME RA-SA-2009 SRs. Table A-3 is addressed in the Table RAI.2-1 columns titled "RG 1.200 Rev. 2 Issue/Position" and "Supporting Requirements Met". RG 1.200 Revision 2 guidance has been considered and as shown in Table RAI.2-1. With the exception of SR IFSN-A16, all IFPRA SR requirements are met at Capability Category I or above.

Conclusions

A gap assessment of the IFPRA model against RG 1.200, Revision 2 in regards to this LAR was performed. The gap assessment found that RG 1.200 Revision 2 requirements are met. This gap analysis included a review of changes to the ASME RA-Sb-2005, Addendum B SRs used for the initial 2009 peer review and the current ASME/ANS standard endorsed by RG 1.200 Revision 2. RG 1.200 Revision 2 describes one acceptable approach for determining whether the technical accuracy of the PRA, in total or among the parts that are used to support an application, is sufficient to provide guidance in the use of the results, such that the PRA can be used in regulatory decision-making for light-water reactors. The gap assessment found that the regulatory guidance found in RG 1.200 Revision 2 and the endorsed standard, are addressed.

All 2009 IF peer review findings were reported in the ILRT License Amendment Request (Table A-1). As noted in Table A-1, all applicable SRs are met at Capability Category I or above except for SR IFSN-A16. The F&O for SR IFSN-A16 is related to small bore piping. Resolution of this F&O, as indicated in Table A-1, would have a negligible impact on the ILRT risk assessment and will not impact the ILRT risk impact conclusions.

TABLE RAI.2-1: INTERNAL FLOOD – ASME/ANS PRA PEER GAP ASSESSMENT

ASME/ANS RA-2009-SA SR (RA-SB-2005 SR IN PARENTHESIS)	CAPABILITY CATEGORY I[(A)]	CAPABILITY CATEGORY II	CAPABILITY CATEGORY III	COMPARISON TO RA-SB-2005, ADDENDA B	RG 1.200 REV. 2 ISSUE/POSITION	SUPPORTING REQUIREMENT MET?
IFPP-A1 (IF-A1)	DEFINE flood areas by dividing the plant into physically separate areas where a flood area is viewed as generally independent of other areas in terms of the potential for internal flooding effects and flood propagation.			No Change	No objection	Met
IFPP-A2 (IF-A1a)	DEFINE flood areas at the level of **buildings or portions thereof from which there would be no propagation to other modeled buildings or portions thereof.**	DEFINE flood areas at the level of **individual rooms or combined rooms/halls for which plant design features exist to restrict flooding.**		No Change	No objection	Met (II/III)
IFPP-A3 (IF-A1b)	For multi-unit sites with shared systems or structures, INCLUDE multi-unit areas, if applicable.			No Change	No objection	Met
IFPP-A4 (IF-A3)	USE plant information sources that reflects the as-built as-operated plant to support development of flood areas.			No change.	No objection	Met
IFPP-A5 (IF-A4)	CONDUCT a plant walkdown(s) to verify the accuracy of information obtained from plant information sources and to obtain or verify: *(a)* spatial information needed for the development of flood areas, and *(b)* plant design features credited in defining flood areas. Note: Walkdown(s) may be done in conjunction with the requirements of IFSO-A6, IFSN-A17, and IFQU-A11.			Editorial change only. References to IF-B3a, IF-C9 and IF-E8 changed to references to of IFSO-A6, IFSN-A17, and IFQU-A11.	No objection	Met

TABLE RAI.2-1: INTERNAL FLOOD – ASME/ANS PRA PEER GAP ASSESSMENT

ASME/ANS RA-2009-SA SR (RA-SB-2005 SR IN PARENTHESIS)	CAPABILITY CATEGORY I[(A)]	CAPABILITY CATEGORY II	CAPABILITY CATEGORY III	COMPARISON TO RA-SB-2005, ADDENDA B	RG 1.200 REV. 2 ISSUE/POSITION	SUPPORTING REQUIREMENT MET?
IFPP-B1 (IF-F2)	DOCUMENT the internal flood plant partitioning in a manner that facilitates PRA applications, upgrades, and peer review. -------------------------------------- RA-Sb-2005 Addenda B IF-F2: DOCUMENT the process used to identify flood sources, flood areas, flood pathways, flood scenarios, and their screening, and internal flood model development and quantification. For example, this documentation typically includes: *(a)* flood sources identified in the analysis, rules used to screen out these sources, and the resulting list of sources to be further examined *(b)* flood areas used in the analysis and the reason for eliminating areas from further analysis *(c)* propagation pathways between flood areas and key assumptions, calculations, or other bases for eliminating or justifying propagation pathways Six additional examples are provided.			IFPP-B1 can be mapped to IF-F2 parts (b) and (c). See Column 2 of this table (Capability Category). Therefore, the Internal Flooding Evaluation is documented in a manner that facilitates PRA applications, upgrades, and peer review.	No objection	Met
IFPP-B2 (IF-F2)	DOCUMENT the process used to identify flood areas. For example, this documentation typically includes: (a) flood areas used in the analysis and the reason for eliminating areas from further analysis (b) any walkdowns performed in support of the plant partitioning -------------------------------------- See RA-Sb-2005 Addenda B IF-F2 description in previous row.			RA-Sb-2005 similar, but provides 9 additional examples. RA-SB-2009 requires documentation of walkdowns in support of plant partitioning, while RA-Sb-2005 is not explicate regarding walkdown documentation. However, this information was necessary to meet SR requirements during the 2009 peer review. Internal Flood Walkdown Notebook (DR PSA-019) documents the process, the impacts, and the features of the model that are important to reproduce the results.	No objection	Met

TABLE RAI.2-1: INTERNAL FLOOD – ASME/ANS PRA PEER GAP ASSESSMENT

ASME/ANS RA-2009-SA SR (RA-SB-2005 SR IN PARENTHESIS)	CAPABILITY CATEGORY I[(A)]	CAPABILITY CATEGORY II	CAPABILITY CATEGORY III	COMPARISON TO RA-SB-2005, ADDENDA B	RG 1.200 REV. 2 ISSUE/POSITION	SUPPORTING REQUIREMENT MET?
IFPP-B3 (IF-F3)	DOCUMENT sources of model uncertainty and related assumptions (as identified in QU-E1 and QU-E2) associated with the internal flood plant partitioning. -------------------------------------- RA-Sb-2005, Addenda B SR IF-F3: Document the sources of model uncertainty and related assumptions (as identified in QU-E1 and QU-E2) associated with the internal flooding analysis.			New SR[(2)]. The RA-Sb-2005 Addenda B SR was a global SR addressing uncertainty and assumptions associated with the internal flood analysis. The internal flood critical assumptions are summarized in the IF notebook. The parametric uncertainty evaluation for the full model incorporates the uncertainty associated with the internal flood modeled sequences. No further characterization of the Internal Flood uncertainty is considered necessary.	No objection	Met
IFSO-A1 (IF-B1)	For each flood area, IDENTIFY the potential sources of flooding [Note (1)]. INCLUDE: *(a)* equipment (e.g., piping, valves, pumps) located in the area that are connected to fluid systems (e.g., circulating water system, service water system, component cooling water system, feedwater system, condensate and steam systems) *(b)* plant internal sources of flooding (e.g., tanks or pools) located in the flood area *(c)* plant external sources of flooding (e.g., reservoirs or rivers) that are connected to the area through some system or structure *(d)* in-leakage from other flood areas (e.g., back flow through drains, doorways, etc.)			No Change	The list of fluid systems should be expanded to include fire protection systems.	Met All significant flood sources identified for evaluation. For example, Fire Protection System contribution is included for Reactor Building (RB) and Turbine Building (TB).
IFSO-A2 (IF-B1a)	For multi-unit sites with shared systems or structures, INCLUDE any potential sources with multi-unit or cross-unit impacts.			No Change	No objection	Met

TABLE RAI.2-1: INTERNAL FLOOD – ASME/ANS PRA PEER GAP ASSESSMENT

ASME/ANS RA-2009-SA SR (RA-SB-2005 SR IN PARENTHESIS)	CAPABILITY CATEGORY I(A)	CAPABILITY CATEGORY II	CAPABILITY CATEGORY III	COMPARISON TO RA-SB-2005, ADDENDA B	RG 1.200 REV. 2 ISSUE/POSITION	SUPPORTING REQUIREMENT MET?
IFSO-A3 (IF-B1b)	SCREEN OUT flood areas with none of the potential sources of flooding listed in IFSO-A1 and IFSO-A2.			Editorial change only. References to SRs IF-B1and IF-B1a changed to references to IFSO-A1 and IFSO-A2.	No objection	Met
IFSO-A4 (IF-B2)	For each potential source of flooding, IDENTIFY the flooding mechanisms that would result in a fluid release. INCLUDE: (a) failure modes of components such as pipes, tanks, gaskets, expansion joints, fittings, seals, etc. (b) human-induced mechanisms that could lead to overfilling tanks, diversion of flow through openings created to perform maintenance; inadvertent actuation of fire suppression system (c) other events resulting in a release into the flood area			No change.	No objection	Met
IFSO-A5 (IF-B3)	For each source and its identified failure mechanism, IDENTIFY the characteristic of release and the capacity of the source. INCLUDE: (a) a characterization of the breach, including type (e.g., leak, rupture, spray) (b) flow rate (c) capacity of source (e.g., gallons of water) (d) the pressure and temperature of the source			No change.	It is necessary to consider a range of flow rates for identified flooding sources, each having a frequency of occurrence. For example, small leaks that only cause spray are more likely than large leaks that may cause equipment submergences.	Met Water release characteristics (e.g., flow rate, capacity, pressure, temperature) of each source identified. Spray and rupture flow rates were considered for flood initiating event frequencies.
IFSO-A6 (IF-B3a)	CONDUCT plant walkdown(s) to verify the accuracy of information obtained from plant information sources and to determine or verify the location of flood sources and in-leakage pathways Note: Walkdown(s) may be done in conjunction with the requirements of IFPP-A5, IFSN-A17, and IFQU-A11.			Editorial change only. References to SRs IF-A4, IF-C9 and IF-E8 changed to references to IFPP-A5, IFSN-A17, and IFQU-A11.	No objection	Met

TABLE RAI.2-1: INTERNAL FLOOD – ASME/ANS PRA PEER GAP ASSESSMENT

ASME/ANS RA-2009-SA SR (RA-SB-2005 SR IN PARENTHESIS)	CAPABILITY CATEGORY I(A)	CAPABILITY CATEGORY II	CAPABILITY CATEGORY III	COMPARISON TO RA-SB-2005, ADDENDA B	RG 1.200 REV. 2 ISSUE/POSITION	SUPPORTING REQUIREMENT MET?
IFSO-B1 (IF-F1)	DOCUMENT the internal flood sources in a manner that facilitates PRA applications, upgrades, and peer review ------------------------------------- RA-Sb-2005 Addenda B SR IF-F1: DOCUMENT the internal flooding analysis in a manner that facilitates PRA applications, upgrades, and peer review.			The RA-Sb-2005 Addenda B SR was a global SR addressing documentation of the internal flood analysis.	No objection	Met
IFSO-B2 (IF-F1)	DOCUMENT the process used to identify applicable flood sources. For example, this documentation typically includes: (a) flood sources identified in the analysis, rules used to screen out these sources, and the resulting list of sources to be further examined (b) screening criteria used in the analysis (c) calculations or other analyses used to support or refine the flooding evaluation (d) any walkdowns performed in support of the identification or screening of flood sources ------------------------------------- RA-Sb-2005 Addenda B SR IF-F1: DOCUMENT the internal flooding analysis in a manner that facilitates PRA applications, upgrades, and peer review.			The RA-Sb-2005 Addenda B SR was a global SR addressing documentation of the internal flood analysis.	No objection	Met
IFSO-B3 (IF-F3)	DOCUMENT sources of model uncertainty and related assumptions (as identified in QU-E1 and QU-E2) associated with the internal flood sources. ------------------------------------- RA-Sb-2005, Addenda B SR IF-F3: Document the sources of model uncertainty and related assumptions (as identified in QU-E1 and QU-E2) associated with the internal flooding analysis.			New SR(2). The RA-Sb-2005 Addenda B SR was a global SR addressing uncertainty and assumptions associated with the internal flood analysis. (See IFPP-B3 (IF-F3).)	No objection	Met
IFSN-A1 (IF-C1)	For each defined flood area and each flood source, IDENTIFY the propagation path from the flood source area to its area of accumulation.			No change.	No objection	Met

TABLE RAI.2-1: INTERNAL FLOOD – ASME/ANS PRA PEER GAP ASSESSMENT

ASME/ANS RA-2009-SA SR (RA-SB-2005 SR IN PARENTHESIS)	CAPABILITY CATEGORY I(A)	CAPABILITY CATEGORY II	CAPABILITY CATEGORY III	COMPARISON TO RA-SB-2005, ADDENDA B	RG 1.200 REV. 2 ISSUE/POSITION	SUPPORTING REQUIREMENT MET?
IFSN-A2 (IF-C2)	For each defined flood area and each flood source, IDENTIFY plant design features that have the ability to terminate or contain the flood propagation. INCLUDE the presence of : (a) flood alarms, (b) flood dikes, curbs, sumps (i.e., physical structures that allow for the accumulation and retention of water), (c) drains (i.e., physical structures that can function as drains), (d) sump pumps, spray shields, water-tight doors, and (e) blowout panels or dampers with automatic or manual operation capability.			No change.	No objection	Met
IFSN-A3 (IF-C2a)	For each defined flood area and each flood source, IDENTIFY those automatic or operator responses that have the ability to terminate or contain the flood propagation.			No change.	No objection	Met
IFSN-A4 (IF-C2b)	ESTIMATE the capacity of the drains and the amount of water retained by sumps, berms, dikes and curbs. ACCOUNT for these factors in estimating flood volumes and SSC impacts from flooding.			No change.	No objection	Met

TABLE RAI.2-1: INTERNAL FLOOD – ASME/ANS PRA PEER GAP ASSESSMENT

ASME/ANS RA-2009-SA SR (RA-SB-2005 SR IN PARENTHESIS)	CAPABILITY CATEGORY I(A)	CAPABILITY CATEGORY II	CAPABILITY CATEGORY III	COMPARISON TO RA-SB-2005, ADDENDA B	RG 1.200 REV. 2 ISSUE/POSITION	SUPPORTING REQUIREMENT MET?
IFSN-A5 (IF-C2c)	For each flood area not screened out using the requirements under other Internal Flooding SRs (e.g., IFSO-A3 and IFSN-A12), IDENTIFY the SSCs located in each defined flood area and along flood propagation paths that are modeled in the internal events PRA model as being required to respond to an initiating event or whose failure would challenge normal plant operation, and are susceptible to flood. For each identified SSC, IDENTIFY, for the purpose of determining its susceptibly per IFSN-A6, its spatial location in the area and any flooding mitigative features (e.g., shielding, flood or spray capability ratings).			Editorial change only. A reference to SR IF-C2c was changed to a reference to IFSN-A6.	No objection	Met
IFSN-A6 (IF-C3)	For the SSCs identified in IFSN-A5, IDENTIFY the susceptibility of each SSC in a flood area to flood-induced failure mechanisms. **INCLUDE failure by submergence and spray** in the identification process. EITHER: (a) ASSESS qualitatively the impact of flood-induced mechanisms that are not formally addressed (e.g., using the mechanisms listed under Capability Category III of this requirement), by using conservative assumptions; OR (b) NOTE that these mechanisms are not included in the scope of the evaluation.		For the SSCs identified in IFSN-A5, IDENTIFY the susceptibility of each SSC in a flood area to flood-induced failure mechanisms. **INCLUDE failure by submergence, spray, jet impingement, pipe whip, humidity, condensation, temperature concerns, and any other identified failure modes** in the identification process.	CAT III requirements added. No change to CAT I-II requirements. The potential for component failure due to flooding induced jet impingement, humidity, condensation, temperature concerns were investigated. Jet impingement and humidity effects from high pressure, high temperature sources are explicitly addressed in one of the following: (a) ISLOCA (b) BOC	No objection to Category I. For Cat II, it is not acceptable to just note that a flood-induced failure mechanism is not included in the scope of the internal flooding analysis. Some level of assessment is required.	Met As noted in the comparison column, RG 1.200 Rev. 2 issue has been addressed.
IFSN-A7 (IF-C3a)	In applying SR IFSN-A6 to determine susceptibility of SSCs to flood-induced failure mechanisms, TAKE CREDIT for the operability of SSCs identified in IFSN-A5 with respect to internal flooding impacts only if supported by an appropriate combination of: (a) test or operational data (b) engineering analysis (c) expert judgment.			Editorial change only. References to SRs IF-C2c and IF-C3 changed to references to IFSN-A6 and IFSN-A5.	No objection	Met

TABLE RAI.2-1: INTERNAL FLOOD – ASME/ANS PRA PEER GAP ASSESSMENT

ASME/ANS RA-2009-SA SR (RA-SB-2005 SR IN PARENTHESIS)	CAPABILITY CATEGORY I[(A)]	CAPABILITY CATEGORY II	CAPABILITY CATEGORY III	COMPARISON TO RA-SB-2005, ADDENDA B	RG 1.200 REV. 2 ISSUE/POSITION	SUPPORTING REQUIREMENT MET?
IFSN-A8 (IF-C3b)	No requirement for inter-area propagation given that flood areas are independent (see SR IFPP-A2).	IDENTIFY inter-area propagation through the normal flow path from one area to another via drain lines; and areas connected via back flow through drain lines involving failed check valves, pipe and cable penetrations (including cable trays), doors, stairwells, hatchways, and HVAC ducts. INCLUDE potential for structural failure (e.g., of doors or walls) due to flooding loads.	IDENTIFY inter-area propagation through the normal flow path from one area to another via drain lines; and areas connected via back flow through drain lines involving failed check valves, pipe and cable penetrations (including cable trays), doors, stairwells, hatchways, and HVAC ducts. INCLUDE potential for structural failure (e.g., of doors or walls) due to flooding loads**, and the potential for barrier unavailability, including maintenance activities.**	Editorial change only. Reference to SR IF-A1a changed to a reference to IFPP-A2.	No objection	Met
IFSN-A9 (IF-C3c)	PERFORM any necessary engineering calculations for flood rate, time to reach susceptible equipment, and the structural capacity of SSCs in accordance with the applicable requirements described in subsection 2.2.3.			No change. Reference to Table 4.5.3-2(b) changed to a reference to subsection 2.2.3.	No objection	Met
IFSN-A10 (IF-C4)	DEVELOP flood scenarios (i.e., the set of information regarding the flood area, source, flood rate and source capacity, operator actions, and SSC damage that together form the boundary conditions for the interface with the internal events PRA) by examining the equipment and relevant plant features in the flood area and areas in potential propagation paths, giving credit for appropriate flood mitigation systems or operator actions, and identifying susceptible SSCs.			No change.	No objection	Met
IFSN-A11 (IF-C4a)	For multi-unit sites with shared systems or structures, INCLUDE multi-unit scenarios.			No change.	No objection	Met

TABLE RAI.2-1: INTERNAL FLOOD – ASME/ANS PRA PEER GAP ASSESSMENT

ASME/ANS RA-2009-SA SR (RA-SB-2005 SR IN PARENTHESIS)	CAPABILITY CATEGORY I(A)	CAPABILITY CATEGORY II	CAPABILITY CATEGORY III	COMPARISON TO RA-SB-2005, ADDENDA B	RG 1.200 REV. 2 ISSUE/POSITION	SUPPORTING REQUIREMENT MET?
IFSN-A12 (IF-C5)	SCREEN OUT ***flood areas*** where flooding of the area does not cause an initiating event or a need for immediate plant shutdown, AND either of the following applies: *(a)* the flood area (including adjacent areas where flood sources can propagate) contains no mitigating equipment modeled in the PRA; OR *(b)* the flood area has no flood sources sufficient (e.g., through spray, immersion, or other applicable mechanism) to cause failure of the equipment identified in IFSN-A5. DO NOT USE failure of a barrier against inter-area propagation to justify screening (i.e., for the purposes of screening, do not credit such failures as a means of beneficially draining the area) JUSTIFY any other qualitative screening criteria.			No change.	No objection	Met
IFSN-A13 (IF-C5a)	SCREEN OUT ***flood areas*** where flooding of the area does not cause an initiating event or a need for immediate plant shutdown, AND the following applies: The flood area contains flooding mitigation systems (e.g., drains or sump pumps) capable of preventing unacceptable flood levels, and the nature of the flood does not cause equipment failure (e.g., through spray, immersion, or other applicable failure mechanisms). DO NOT CREDIT mitigation systems for screening out flood areas unless there is a definitive basis for crediting the capability and reliability of the flood mitigation system(s).			No change.	No objection	Met

TABLE RAI.2-1: INTERNAL FLOOD – ASME/ANS PRA PEER GAP ASSESSMENT

<table>
<tr><th>ASME/ANS RA-2009-SA SR (RA-SB-2005 SR IN PARENTHESIS)</th><th>CAPABILITY CATEGORY I[(A)]</th><th>CAPABILITY CATEGORY II</th><th>CAPABILITY CATEGORY III</th><th>COMPARISON TO RA-SB-2005, ADDENDA B</th><th>RG 1.200 REV. 2 ISSUE/POSITION</th><th>SUPPORTING REQUIREMENT MET?</th></tr>
<tr><td>IFSN-A14 (IF-C6)</td><td>USE potential human mitigative actions as additional criteria for screening out flood areas if all the following can be shown:
(a) flood indication is available in the control room
(b) the flood sources in the area can be isolated
(c) the time to the damage of safe shutdown equipment is significantly greater than the expected time for human mitigative actions to be performed, for the worst flooding initiator.</td><td>USE potential human mitigative actions as additional criteria for screening out flood areas if all the following can be shown:
(a) flood indication is available in the control room
(b) the flood sources in the area can be isolated
(c) the mitigative action can be performed with high reliability for the worst flooding initiator. High reliability is established by demonstrating, for example, that the actions are procedurally directed, that adequate time is available for response, that the area is accessible, and that there is sufficient manpower available to perform the actions.</td><td>DO NOT SCREEN OUT flood areas based on reliance on operator action to prevent challenges to normal plant operations.</td><td>No change.</td><td>No objection</td><td>Met (III)</td></tr>
<tr><td>IFSN-A15 (IF-C7)</td><td colspan="3">SCREEN OUT flood sources if it can be shown that:
(a) the flood source is insufficient (e.g., through spray, immersion, or other applicable mechanism) to cause failure of equipment identified in IFSN-A5; OR
(b) the area flooding mitigation systems (e.g., drains or sump pumps) are capable of preventing unacceptable flood levels and nature of the flood does not cause failure of equipment identified in IFSN-A5 (e.g., through spray, immersion, or other applicable failure mechanism); OR
(c) the flood only affects the system that is the flood source and the systems analysis addresses this per SY-A13 and SY-A14 and need not be treated as a separate internal flooding initiating event.</td><td>No change.</td><td>No objection</td><td>Met</td></tr>
</table>

TABLE RAI.2-1: INTERNAL FLOOD – ASME/ANS PRA PEER GAP ASSESSMENT

ASME/ANS RA-2009-SA SR (RA-SB-2005 SR IN PARENTHESIS)	CAPABILITY CATEGORY I[(A)]	CAPABILITY CATEGORY II	CAPABILITY CATEGORY III	COMPARISON TO RA-SB-2005, ADDENDA B	RG 1.200 REV. 2 ISSUE/POSITION	SUPPORTING REQUIREMENT MET?
IFSN-A16 (IF-C8)	USE potential human mitigative actions as additional criteria for screening out ***flood sources*** if all the following can be shown: *(a)* flood indication is available in the control room, *(b)* the flood source can be isolated, and ***(c)* the time to the damage of safe shutdown equipment is significantly greater than the expected time for human mitigative actions to be performed, for the worst flood from that source.**	USE potential human mitigative actions as additional criteria for screening out ***flood sources*** if all the following can be shown: *(a)* flood indication is available in the control room, *(b)* the flood source can be isolated, and *(c)* the mitigative action can be performed with high reliability for the worst flood from that source. High reliability is established by demonstrating, for example, that the actions are procedurally directed, that adequate time is available for response, that the area is accessible, and that th**ere is sufficient manpower available to perform the actions.**	**DO NOT SCREEN OUT *flood sources* based on reliance on operator action to prevent challenges to normal plant operations.**	No change.	No objection	Not Met. See Table A-1 of the ILRT submittal for rationale for having no impact to the ILRT application.
IFSN-A17 (IF-C9)	CONDUCT plant walkdown(s) to verify the accuracy of information obtained from plant information sources and to obtain or verify: *(a)* SSCs located within each defined flood area *(b)* flood / spray / other applicable mitigative features of the SSCs located within each defined flood area (e.g., drains, shields, etc.) *(c)* pathways that could lead to transport to the flood area Note: Walkdown(s) may be done in conjunction with the requirements of IFPP-A5, IFSO0A6, and IFQU-A11.			Editorial change only. References to SRs IF-A4, IF-B3a and IF-E8 changed to references to of IFPP-A5, IFSO0A6, and IFQU-A11.	No objection	Met

TABLE RAI.2-1: INTERNAL FLOOD – ASME/ANS PRA PEER GAP ASSESSMENT

ASME/ANS RA-2009-SA SR (RA-SB-2005 SR IN PARENTHESIS)	CAPABILITY CATEGORY I[(A)]	CAPABILITY CATEGORY II	CAPABILITY CATEGORY III	COMPARISON TO RA-SB-2005, ADDENDA B	RG 1.200 REV. 2 ISSUE/POSITION	SUPPORTING REQUIREMENT MET?
IFSN-B1 (IF-F1)	DOCUMENT the internal flood scenarios in a manner that facilitates PRA applications, upgrades, and peer review. -------------------------------------- RA-Sb-2005 Addenda B SR IF-F1: DOCUMENT the internal flooding analysis in a manner that facilitates PRA applications, upgrades, and peer review.			The RA-Sb-2005 Addenda B SR was a global SR addressing documentation of the internal flood analysis.	No objection	Met
IFSN-B2 (IF-F2)	DOCUMENT the process used to identify applicable flood scenarios. For example, this documentation typically includes: (a) propagation pathways between flood areas and assumptions, calculations, or other bases for eliminating or justifying propagation pathways (b) accident mitigating features and barriers credited in the analysis, the extent to which they were credited, and associated justification (c) assumptions or calculations used in the determination of the impacts of submergence, spray, temperature, or other flood-induced effects on equipment operability (d) screening criteria used in the analysis (e) flooding scenarios considered, screened, and retained (f) description of how the internal event analysis models were modified to model these remaining internal flood scenarios (g) calculations or other analyses used to support or refine the flooding evaluation (h) any walkdowns performed in support of the identification or screening of flood scenarios			Insignificant change. Items (a) through (g) match IF-F2 requirements. Item (h) is not covered under IF-F2, however, walkdowns performed in support of identification of flood scenarios are captured in the Internal Flood Walkdown Notebook.	No objection	Met

TABLE RAI.2-1: INTERNAL FLOOD – ASME/ANS PRA PEER GAP ASSESSMENT

ASME/ANS RA-2009-SA SR (RA-SB-2005 SR IN PARENTHESIS)	CAPABILITY CATEGORY I[(A)]	CAPABILITY CATEGORY II	CAPABILITY CATEGORY III	COMPARISON TO RA-SB-2005, ADDENDA B	RG 1.200 REV. 2 ISSUE/POSITION	SUPPORTING REQUIREMENT MET?
IFSN-B3 (IF-F3)	DOCUMENT sources of model uncertainty and related assumptions (as identified in QU-E1 and QU-E2) associated with the internal flood scenarios. ------------------------------------- RA-Sb-2005, Addenda B SR IF-F3: Document the sources of model uncertainty and related assumptions (as identified in QU-E1 and QU-E2) associated with the internal flooding analysis.			New SR[(2)]. The RA-Sb-2005 Addenda B SR was a global SR addressing uncertainty and assumptions associated with the internal flood analysis. (See IFPP-B3 (IF-F3).)	No objection	Met
IFEV-A1 (IF-D1)	For each flood scenario, IDENTIFY the corresponding plant initiating event group identified per subsection 2.2.1 and the scenario-induced failures of SSCs required to respond to the plant initiating event. INCLUDE the potential for a flooding-induced transient or LOCA. If an appropriate plant initiating event group does not exist, CREATE a new plant initiating event group in accordance with the applicable requirements of subsection 2.2.1.			Editorial change only. Reference to Table 4.5.1-2(b) changed to reference to subsection 2.2.1.	No objection	Met

TABLE RAI.2-1: INTERNAL FLOOD – ASME/ANS PRA PEER GAP ASSESSMENT

ASME/ANS RA-2009-SA SR (RA-SB-2005 SR IN PARENTHESIS)	CAPABILITY CATEGORY I(A)	CAPABILITY CATEGORY II	CAPABILITY CATEGORY III	COMPARISON TO RA-SB-2005, ADDENDA B	RG 1.200 REV. 2 ISSUE/POSITION	SUPPORTING REQUIREMENT MET?
IFEV-A2 (IF-D3)	GROUP flooding scenarios identified in IFSN-A10 only when the following is true: *(a)* scenarios can be considered similar in terms of plant response, success criteria, timing, and the effect on the operability and performance of operators and relevant mitigating systems; or *(b)* scenarios can be subsumed into a group and bounded by the worst-case impacts within the "new" group.	GROUP flooding scenarios identified in IFSN-A10 only when the following is true: *(a)* scenarios can be considered similar in terms of plant response, success criteria, timing, and the effect on the operability and performance of operators and relevant mitigating systems; or *(b)* scenarios can be subsumed into a group and bounded by the worst case impacts within the "new" group. **AVOID subsuming scenarios into a group unless:** ***(i)* the impacts are comparable to or less than those of the remaining scenarios in that group,** **AND** ***(ii)* it is demonstrated that such grouping does not impact significant accident sequences.**	GROUP flooding scenarios identified in IFSN-A10 only when the following is true: *(a)* scenarios can be considered similar in terms of plant response, success criteria, timing, and the effect on the operability and performance of operators and relevant mitigating systems; or *(b)* scenarios can be subsumed into a group and bounded by the worst case impacts within the "new" group. **DO NOT ADD scenarios to a group and DO NOT SUBSUME scenarios into a group unless the impacts are comparable to those of the remaining scenarios in that group.**	Editorial change only. Reference to IF-C4 changed to a reference to IFSN-A10.	No objection	Met (II)
IFEV-A3 (IF-D3a)	GROUP OR SUBSUME the flood initiating scenarios with an existing plant initiating event group, if the impact of the flood (i.e., plant response and mitigating system capability) is the same as a plant initiating event group already considered in the PRA in accordance with the applicable requirements of subsection 2.2.1.		DO NOT GROUP AND DO NOT SUBSUME flood initiating scenarios with other plant initiating event groups.	Editorial change only. Reference to Table 4.5.1-2(b) changed to a reference to subsection 2.2.1.	No objection	Met

TABLE RAI.2-1: INTERNAL FLOOD – ASME/ANS PRA PEER GAP ASSESSMENT

ASME/ANS RA-2009-SA SR (RA-SB-2005 SR IN PARENTHESIS)	CAPABILITY CATEGORY I(A)	CAPABILITY CATEGORY II	CAPABILITY CATEGORY III	COMPARISON TO RA-SB-2005, ADDENDA B	RG 1.200 REV. 2 ISSUE/POSITION	SUPPORTING REQUIREMENT MET?
IFEV-A4 (IF-D4)	For multi-unit sites with shared systems or structures, INCLUDE multi-unit impacts on SSCs and plant initiating events caused by internal flood scenario groups.			No change.	No objection	Met
IFEV-A5 (IF-D5)	DETERMINE the flood initiating event frequency for each flood scenario group by using the applicable requirements in subsection 2.2.1.			Editorial change only. Reference to Table 4.5.1-2(c) changed to a reference to subsection 2.2.1.	No objection	Met
IFEV-A6 (IF-D5a)	In determining the flood initiating event frequencies for flood scenario groups, USE **one of the following**: (a) generic operating experience (b) pipe, component, and tank rupture failure rates from generic data sources (c) **a combination of (a) or (b) above with** engineering judgment.	**GATHER plant-specific information on plant design, operating practices and conditions that may impact flood likelihood (i.e., material condition of fluid systems, experience with water hammer, and maintenance induced floods).** In determining the flood initiating event frequencies for flood scenario groups, USE **a combination of** (a) generic **and plant-specific** operating experience, (b) pipe, component, and tank rupture failure rates from generic data sources **and plant-specific** experience		No change to Category I. Engineering judgement item c removed from Category II. No impact as ILRT is Category I application and DNPS does not have any significant internal flood events in their history. There have been no applicable plant specific flood events identified that would alter the generic frequencies.	No objection	Met (II/III)
IFEV-A7 (IF-D6)	INCLUDE consideration of human-induced floods during maintenance through application of generic data.		EVALUATE plant-specific maintenance activities for potential human-induced floods using human reliability analysis techniques. **NOTE: This would require consideration of errors of commission. Subsection 2-2.5 does not at this time provide specific requirements related to errors of commission.**	Editorial change only. Reference to Table 4.5.5 changed to a reference to subsection 2.2-5.	No objection	Met

TABLE RAI.2-1: INTERNAL FLOOD – ASME/ANS PRA PEER GAP ASSESSMENT

ASME/ANS RA-2009-SA SR (RA-SB-2005 SR IN PARENTHESIS)	CAPABILITY CATEGORY I(A)	CAPABILITY CATEGORY II	CAPABILITY CATEGORY III	COMPARISON TO RA-SB-2005, ADDENDA B	RG 1.200 REV. 2 ISSUE/POSITION	SUPPORTING REQUIREMENT MET?
IFEV-A8 (IF-D7)	SCREEN OUT flood scenario groups if (a) the quantitative screening criteria in IFSN-A10, as applied to the flood scenario groups, are met, OR (b) the internal flooding initiating event affects only components in a single system, AND it can be shown that the product of: • the frequency of the flood and • the probability of SSC failure given the flood is two orders of magnitude lower than: • the product of the non-flooding frequency for the corresponding initiating event in the PRA and • the random (non-flood-induced) failure probability of the same SSCs that are assumed failed by the flood. If the flood impacts multiple systems, DO NOT screen on this basis.			Editorial change only. Reference to Table IE-C4 changed to a reference to IFSN-A10.	No objection	Met

TABLE RAI.2-1: INTERNAL FLOOD – ASME/ANS PRA PEER GAP ASSESSMENT

ASME/ANS RA-2009-SA SR (RA-SB-2005 SR IN PARENTHESIS)	CAPABILITY CATEGORY I(A)	CAPABILITY CATEGORY II	CAPABILITY CATEGORY III	COMPARISON TO RA-SB-2005, ADDENDA B	RG 1.200 REV. 2 ISSUE/POSITION	SUPPORTING REQUIREMENT MET?
IFEV-B1 (IF-F1)	DOCUMENT the internal flood-induced initiating events in a manner that facilitates PRA applications, upgrades, and peer review. ------------------------------------- RA-Sb-2005 Addenda B SR IF-F1: DOCUMENT the internal flooding analysis in a manner that facilitates PRA applications, upgrades, and peer review.			The RA-Sb-2005 Addenda B SR was a global SR addressing documentation of the internal flood analysis.	No objection	Met
IFEV-B2 (IF-F2)	DOCUMENT the process used to identify applicable flood-induced initiating events. For example, this documentation typically includes (a) flood frequencies, component unreliabilities/unavailabilities, and HEPs used in the analysis (i.e., the data values unique to the flooding analysis) (b) calculations or other analyses used to support or refine the flooding evaluation (c) screening criteria used in the analysis ------------------------------------- RA-Sb-2005 Addenda B IF-F2: DOCUMENT the process used to identify flood sources, flood areas, flood pathways, flood scenarios, and their screening, and internal flood model development and quantification. For example, this documentation typically includes: (f) screening criteria used in the analysis (i) flood frequencies, component unreliabilities/unavailabilities, and HEPs used in the analysis (i.e., the data values unique to the flooding analysis) (j) calculations or other analyses used to support or refine the flooding evaluation			New SR, however, requirements overlap with requirements found in SR IF-F2. The Internal Flood Evaluation Notebook (DR PSA-012) and the Internal Flood Walkdown Notebook document the process, the impacts, and the features of the model that are important to reproduce the results. Documentation includes the examples (a), (b) and (c).	No objection	Met

TABLE RAI.2-1: INTERNAL FLOOD – ASME/ANS PRA PEER GAP ASSESSMENT

ASME/ANS RA-2009-SA SR (RA-SB-2005 SR IN PARENTHESIS)	CAPABILITY CATEGORY I(A)	CAPABILITY CATEGORY II	CAPABILITY CATEGORY III	COMPARISON TO RA-SB-2005, ADDENDA B	RG 1.200 REV. 2 ISSUE/POSITION	SUPPORTING REQUIREMENT MET?
IFEV-B3 (IF-F3)	Document sources of model uncertainty and related assumptions (as identified in QU-E1 and QU-E2) associated with the internal flood-induced initiating events. -------------------------------------- RA-Sb-2005, Addenda B SR IF-F3: Document the sources of model uncertainty and related assumptions (as identified in QU-E1 and QU-E2) associated with the internal flooding analysis.			New SR(2). The RA-Sb-2005 Addenda B SR was a global SR addressing uncertainty and assumptions associated with the internal flood analysis. (See IFPP-B3 (IF-F3).)	No objection	Met
IFQU-A1 (IF-E1)	For each flood scenario, REVIEW the accident sequences for the associated plant initiating event group to confirm applicability of the accident sequence model. If appropriate accident sequences do not exist, MODIFY sequences as necessary to account for any unique flood-induced scenarios and/or phenomena in accordance with the applicable requirements described in subsection 2.2.2.			Editorial change only. Reference to paragraph 4.5.2 changed to a reference to subsection 2.2.2.	No objection	Met
IFQU-A2 (IF-E3)	MODIFY the systems analysis results obtained by following the applicable requirements described in subsection 2.2.4 to include flood-induced failures identified by IFSN-A6.			Editorial change only. References to paragraph 4.5.4 and IF-C3 changed to a references to subsection 2.2.2 and IFSN-A6.	No objection	Met
IFQU-A3 (IF-E3a)	SCREEN OUT a flood area if the product of the sum of the frequencies of the flood scenarios for the area and the bounding conditional core damage probability (CCDP) is less than 10^{-9}/reactor year. The bounding CCDP is the highest of the CCDP values for the flood scenarios in an area.		LIMIT THE USE OF quantitative screening of flood areas.	No change.	No objection	Met (I/II)
IFQU-A4 (IF-E4)	If additional analysis of SSC data is required to support quantification of flood scenarios, PERFORM the analysis in accordance with the applicable requirements described in subsection 2.2.6.			Editorial change only. Reference to paragraph 4.5.6 changed to a reference to subsection 2.2.6.	No objection	Met
IFQU-A5 (IF-E5)	If additional human failure events are required to support quantification of flood scenarios, PERFORM any human reliability analysis in accordance with the applicable requirements described in subsection 2.2.5.			Editorial change only. Reference to Tables 4.5.5-2(e) through Table 4.5.5-2(h) changed to a reference to subsection 2.2.5.	No objection	Met

TABLE RAI.2-1: INTERNAL FLOOD – ASME/ANS PRA PEER GAP ASSESSMENT

ASME/ANS RA-2009-SA SR (RA-SB-2005 SR IN PARENTHESIS)	CAPABILITY CATEGORY I[A]	CAPABILITY CATEGORY II	CAPABILITY CATEGORY III	COMPARISON TO RA-SB-2005, ADDENDA B	RG 1.200 REV. 2 ISSUE/POSITION	SUPPORTING REQUIREMENT MET?
IFQU-A6 (IF-E5a)	For all human failure events in the internal flood scenarios, INCLUDE the following scenario-specific impacts on PSFs for control room and ex-control room actions as appropriate to the HRA methodology being used: (a) additional workload and stress (above that for similar sequences not caused by internal floods) (b) cue availability (c) effect of flood on mitigation, required response, timing, and recovery activities (e.g., accessibility restrictions, possibility of physical harm) (d) flooding-specific job aids and training (e.g., procedures, training exercises)			No change.	No objection	Met
IFQU-A7 (IF-E6)	PERFORM internal flood sequence quantification in accordance with the applicable requirements described in subsection 2.2.7.			Editorial change only. Reference to paragraph 4.5.8 changed to a reference to subsection 2.2.7.	No objection	Met
IFQU-A8 (IF-E6a)	INCLUDE, in the quantification, the combined effects of failures caused by flooding and those coincident with the flooding due to independent causes including equipment failures, unavailability due to maintenance, and other credible causes.			No change.	The quantification also needs to include the effect of common-cause failure.	Met The quantification incorporates all of the cited combined effects including the common cause failure effects.
IFQU-A9 (IF-E6b)	INCLUDE, in the quantification, both the direct effects of the flood (e.g., loss of cooling from a service water train due to an associated pipe rupture) and indirect effects such as submergence, jet impingement, and pipe whip, as applicable.			No change.	No objection	Met

TABLE RAI.2-1: INTERNAL FLOOD – ASME/ANS PRA PEER GAP ASSESSMENT

ASME/ANS RA-2009-SA SR (RA-SB-2005 SR IN PARENTHESIS)	CAPABILITY CATEGORY I(A)	CAPABILITY CATEGORY II	CAPABILITY CATEGORY III	COMPARISON TO RA-SB-2005, ADDENDA B	RG 1.200 REV. 2 ISSUE/POSITION	SUPPORTING REQUIREMENT MET?
IFQU-A10 (IF-E7)	For each flood scenario, REVIEW the LERF analysis to confirm applicability of the LERF sequences. If appropriate LERF sequences do not exist, MODIFY the LERF analysis as necessary to account for any unique flood-induced scenarios or phenomena in accordance with the applicable requirements described in subsection 2.2.8.			Editorial change only. Reference to paragraph 4.5.9 changed to a reference to subsection 2.2.8.	No objection	Met
IFQU-A11 (IF-E8)	CONDUCT walkdown(s) to verify the accuracy of information obtained from plant information sources and to obtain or verify inputs to: (a) engineering analyses (b) human reliability analyses (c) spray or other applicable impact assessments (d) screening decisions Note: Walkdown(s) may be done in conjunction with the requirements of IFPP-A5, IFSO-A6, and IFSN-A17.			Editorial change only. References to IF-A4, IF-B3a, and IF-C9 changed to references to IFPP-A5, IFSO-A6, and IFSN-A17.	No objection	Met
IFQU-B1 (IF-F1)	DOCUMENT the internal flood accident sequences and quantification in a manner that facilitates PRA applications, upgrades, and peer review. ------------------------------------- RA-Sb-2005 Addenda B SR IF-F1: DOCUMENT the internal flooding analysis in a manner that facilitates PRA applications, upgrades, and peer review.			The RA-Sb-2005 Addenda B SR was a global SR addressing documentation of the internal flood analysis.	No objection	Met

TABLE RAI.2-1: INTERNAL FLOOD – ASME/ANS PRA PEER GAP ASSESSMENT

ASME/ANS RA-2009-SA SR (RA-SB-2005 SR IN PARENTHESIS)	CAPABILITY CATEGORY I(A)	CAPABILITY CATEGORY II	CAPABILITY CATEGORY III	COMPARISON TO RA-SB-2005, ADDENDA B	RG 1.200 REV. 2 ISSUE/POSITION	SUPPORTING REQUIREMENT MET?
IFQU-B2 (IF-F2)	DOCUMENT the process used to define the applicable internal flood accident sequences and their associated quantification. For example, this documentation typically includes: (a) calculations or other analyses used to support or refine the flooding evaluation (b) screening criteria used in the analysis (c) flooding scenarios considered, screened, and retained (d) results of the internal flood analysis, consistent with the quantification requirements provided in HLR QU-D (e) any walkdowns performed in support of internal flood accident sequence quantification			Items (a) through (d) were contained in IF-F2 requirements. Item (e) addressed in IF Walkdown Notebook. The Internal Flood Evaluation Notebook (DR PSA-012) and the Internal Flood Walkdown Notebook document the process, the impacts, and the features of the model that are important to reproduce the results.	No objection	Met
IFQU-B3 (IF-F3)	DOCUMENT sources of model uncertainty and related assumptions (as identified in QU-E1 and QU-E2) associated with the internal flood accident sequences and quantification. -------------------------------------- RA-Sb-2005, Addenda B SR IF-F3: Document the sources of model uncertainty and related assumptions (as identified in QU-E1 and QU-E2) associated with the internal flooding analysis.			New SR[2]. The RA-Sb-2005 Addenda B SR was a global SR addressing uncertainty and assumptions associated with the internal flood analysis. (See IFPP-B3 (IF-F3).)		Met

Notes to Table RAI.2-1:

(A) Differences in requirements for meeting Capability Categories may exist between ASME/ANS RA-2009-Sa and RA-SB-2005. If there are differences, both sets of requirements are shown, separated by dashed lines – with ASME/ANS RA-2009-Sa requirements listed above the dashed lines, and RA-SB-2005 requirements listed below the dashed lines.

(1) Sources of flooding are typically expected to be water sources. The requirements are generally written in terms of sources of waters, but other fluid sources should also be considered.

(2) The modeling uncertainty treatment has been clarified by the NRC and EPRI via the issuance of NUREG-1855 in February/March of 2009. The ASME/ANS Combined Standard reflects this guidance with revised Supporting Requirements. The following SRs describe these requirements:

- QU-E1: IDENTIFY sources of model uncertainty.
- QU-E2: IDENTIFY assumptions made in the development of the PRA model.
- QU-E4: For each source of model uncertainty and related assumption identified in QU-E1 and QU-E2, respectively, IDENTIFY how the PRA model is affected (e.g., introduction of a new basic event, changes to basic event probabilities, change in success criterion, introduction of a new initiating event).
- QU-F4: DOCUMENT the characterization of the sources of model uncertainty and related assumptions (as identified in QU-E4).
- LE-F3: IDENTIFY and CHARACTERIZE the LERF sources of model uncertainty and related assumptions, consistent with the requirements of Table 3.8-1 (SRs for HLR-QU-D and HLR-QU-E).
- SC-C3, SY-C3, HR-I3, DA-E3, LE-G4, IFPP-B3, IFSO-B3, IFSN-B3, IFEV-B3, and IFQU-B3: DOCUMENT the sources of model uncertainty and related assumptions (as identified in QU-E1 and QU-E2 [or LE-F3]) associated with [each element].

<u>PRA-RAI-3</u> Resolution of PRA Peer Review F&Os (part a)

In Appendix A, "PRA Technical Adequacy," Table A-2, "2016 FPIE Peer Review Findings and Impact to Application," of Attachment 3 (beginning on page A-17), the licensee presents the F&Os from the 2016 peer review of the IEPRA along with their resolutions for this LAR.

a. F&O 2-12 (pg. A-22) pertaining to large early release frequency (LERF), states that credit for mitigating actions by operators, fission product scrubbing, and expected beneficial failures in significant accident progression sequences are included in the LERF analysis, but the licensee provided no justification for equipment survivability and successful human actions under adverse environments or after containment failure. In addition, the resolution states that Capability Category I (CC-I) is met.

The NRC staff requests that the licensee explain how CC-I is satisfied.

<u>EGC Response (part a):</u>

Credit for survivability of equipment and human actions under adverse environments was modeled with basic event 2MUPH-ENVIR--F-- in the Containment Event Tree (CET) node for Long-Term Coolant Inventory Makeup (MU). The MU node evaluates the continued availability of water injection to the containment and/or RPV following containment failure for equipment located inside secondary containment that may be subject to a harsh environment given containment failure. This basic event models minimal credit (0.9 failure probability) with a documented basis of engineering judgment.

In consideration of the ILRT risk assessment, the ILRT model was re-run with 2MUPH-ENVIR--F-- set to 1.0. There was no change to the CDF or LERF results at a truncation limit of 1E-11/year as shown in Table RAI.3-1.

TABLE RAI.3-1: ENVIRONMENT SENSITIVITY RESULTS

	FPIE LEVEL 1		**FPIE LEVEL 2**	
	# OF CUTSETS	**CDF(1/YR)**	**# OF CUTSETS**	**LERF(1/YR)**
2013A Base Model	6293	3.2181E-6	3078	7.3052E-07
2MUPH-ENVIR--F-- = 1.0	6293	3.2181E-6	3078	7.3052E-07

Based on the confirmation of non-quantitative impact on the ILRT model, F&O 2-12 is found to have no impact on the ILRT risk assessment.

<u>PRA-RAI-3</u> Resolution of PRA Peer Review F&Os (part b)

b. F&O 5-1 (pg. A-25) pertains to the modeling of human reliability analysis (HRA) dependencies. The peer review team found that the "Dresden PRA model does not account for the influence of success or failure in preceding human actions and system performance on the human event under consideration," including timing, factors that could lead to dependence, and availability of resources and therefore SRs HR-G7 and QU-C2 were found Not Met. In the "Impact to Application" column of the table, the licensee however states that the HRA dependency methodology is considered adequate for this application.

The NRC staff requests that the licensee either provide the results of a focused scope peer review on the HRA dependency analysis to demonstrate that SRs HR-G7 and QU-C2 are met, or, alternatively provide the results of a sensitivity study, which applies the minimum joint human error probability (HEP) floor value of 1E-6 for internal event PRAs.

<u>EGC Response (part b):</u>

A sensitivity case was performed applying the minimum joint human error probability (HEP) floor value of 1E-6 for internal events. This sensitivity required a change to two recovery HEPs from 5E-07 to 1E-06. These recovery events applied only to long term decay heat removal sequences (Class II and IIE). The baseline CDF value increased by 6.3E-08/year. This is an increase of only 1.95%. Given the small increase in CDF and the low likelihood of a Large Early Release from a Class II event, there is a negligible impact to ILRT LERF results.

TABLE RAI.3-2: MINIMUM JOINT HEP FLOOR SENSITIVITY

	FPIE LEVEL 1 CDF(1/YR)	**FPIE LEVEL 2 LERF(1/YR)**
2013A Base Model	3.218E-6	7.3052E-07
Recovery HEPs at 1E-6	3.281E-6	7.3063E-07
Delta Risk	6.3E-8 (1.95% increase)	1.1E-10 (0.02% increase)

***PRA-RAI-3* Resolution of PRA Peer Review F&Os (part c)**

In Section 3.4.2.5, Table A-2, of Attachment 3, the licensee stated that five SRs were assessed as not applicable to the DNPS PRA. The results of that assessment are used as the basis for the capability assessment provided in the LAR.

c. The NRC staff requests the licensee to provide greater specificity regarding the five SRs mentioned in the above reference and their relevance to this LAR.

EGC Response (part c):

The five SRs assessed as not applicable to the DNPS PRA by the peer review team and their relevance to the ILRT LAR are listed in Table RAI.3-3 below.

TABLE RAI.3-3: RELEVANCE OF SRS NOT APPLICABLE TO THE DNPS PRA

SR	SR REQUIREMENT	RELEVANCE TO DNPS ILRT
AS-B4	When the event trees with conditional split fraction method are used, if the probability of Event B is dependent on the occurrence or non-occurrence of Event A, where practical, PLACE Event A to the left of Event B in the ordering of event tops. Where not practical, PROVIDE the rationale used for the ordering.	The DNPS PRA does not use the conditional split fraction method. Therefore, this SR is not applicable to the DNPS ILRT LAR.
LE-D5	Capability Category I is the following: "Use a conservative evaluation of secondary side isolation capability for significant accident progression sequences caused by SG tube failure resulting in a large early release. If generic analyses generated for a similar plants are used…."	The LE-D5 requirements apply to plants (PWRs) with steam generators. DNPS is a BWR, without steam generators. Therefore, this SR is not applicable to the DNPS ILRT LAR.
LE-D6	Capability Category I is the following: "Use a conservative analysis of thermally induced SG tube rupture that includes plant-specific procedures. An acceptable alternative is the approach in NUREG/CR-6595…"	The LE-D6 requirements apply to plants (PWRs) with steam generators. DNPS is a BWR, without steam generators. Therefore, this SR is not applicable to the DNPS ILRT LAR.

TABLE RAI.3-3: RELEVANCE OF SRS NOT APPLICABLE TO THE DNPS PRA

SR	SR REQUIREMENT	RELEVANCE TO DNPS ILRT
QU-B10	If modules, subtrees, or split fractions are used to facilitate the quantification, USE a process that allows: (a) identification of shared events (b) correct formation of modules that are truly independent (c) results interpretation based on individual events within modules (e.g., risk significance)	QU-B10 requirements apply to PRA that use modules, subtrees, or split fractions during quantification. The DNPS PRA does not use these elements. Therefore, this SR is not applicable to the DNPS ILRT LAR. (Note, this SR applies to RiskMan® model structures; DNPS does not employ those model structures.)
SY-A9	If a system model is developed in which a single failure of a super component (or module) is used to represent the collective impact of failures of several components, PERFORM the modularization process in a manner that avoids grouping events with different recovery potential, events that are required by other systems, or events that have probabilities that are dependent on the scenario. Examples of such events include: (a) hardware failures that are not recoverable versus actuation signals, which are recoverable (b) HE events that can have different probabilities dependent on the context of different accident sequences (c) events that are mutually exclusive of other events not in the module (d) events that occur in other fault trees (especially common-cause events) (e) SSCs used by other systems	The DNPS PRA does not modularize. Therefore, the issues identified in SY-A9 are not applicable to the DNPS PRA model. It is noted that the RPS mechanical and electrical component failures are modeled as super components based on data from NUREG 5500 Volume III. This treatment is not considered to equivalent to modularization as exemplified by (a) through (e) of this SR.

Greater specificity regarding the SRs determined by the peer reviewers as not applicable to the DNPS PRA is provided in the table above. As these SRs are not applicable to the DNPS PRA, they have no relevance to the DNPS ILRT LAR.

***PRA-RAI-4* PRA Differences (part a)**

In Section 4.2, "Plant Specific Inputs," of Attachment 3 (page 4-8), the licensee states that no substantive differences exist between DNPS, Unit 2 and Unit 3 that are judged to affect the conclusions of the PRA, as such, no separate PRA quantification is conducted for DNPS Unit 3.

The NRC staff requests that the licensee:

a. *Provide clarification as to whether this conclusion also applies to the IFPRA model, and, if not, discuss the impact of any risk-significant differences.*

EGC Response (part a):

The IFPRA is included in the Unit 2 and Unit 3 Internal Events PRA models of record referenced in Section 4.2 "Plant Specific Inputs". A review of the internal flood analysis documentation and PRA modeling found that no substantive differences exist between the DNPS Unit 2 and Unit 3 that are judged to affect the conclusions of the PRA. The internal flooding analysis covered both units and considered the impact of flood sources located in the opposite unit. Unit differences are minimal, resulting in an insignificant difference between internal flood CDF contributions. Therefore, it is appropriate to use the Unit 2 FPIE results, including the internal flood contribution, in determining the ILRT impact to Unit 3.

A discussion of FPIE results, plant differences and unit interfaces pertinent to IFPRA modeling follow:

- In the 2013 DNPS PRA, the total contribution to CDF from internal flooding for both Unit 2 and Unit 3 is 5.1% of CDF.
- The Unit 2 and Unit 3 base quantification results for CDF and LERF are the same to 3 significant digits and therefore, the IFPRA CDF contribution of 5.1% is approximately the same for each unit.

		2013 Unit 2 DNPS Model	2013 Unit 3 DNPS Model
CDF	=	3.22E-6/yr	3.22E-6/yr
LERF	=	7.31E-7/yr	7.31E-7/yr

- Based on similar pipe lengths and routing, the internal flooding initiating events found in the base cutsets for both Unit 2 and Unit 3 use the same frequency (e.g., %FLFPTB ‘INIT: FPS & DGCW RUPTURE IN RB’ uses a frequency of 5.54E-04/year for both the Unit 2 and Unit 3 PRA models).
- The DNPS Self-Assessment and Internal Flood peer review found multi-unit ASME/ANS Standard Requirements to be "Met" for the DNPS PRA model. These requirements include the following:

- IFPP-A3 CAT I-III: For multi-unit sites with shared systems or structures, INCLUDE multi-unit areas, if applicable.
- IFSO-A2 CAT I-III: For multi-unit sites with shared systems or structures, INCLUDE any potential sources with multi-unit or cross-unit impacts.
- IFSN-A11 CAT I-III: For multi-unit sites with shared systems or structures, INCLUDE multi-unit scenarios.
- IFEF-A4 CATE III: For multi-unit sites with shared systems or structures, INCLUDE multi-unit impacts on SSCs and plant initiating events caused by internal flood scenario groups.

• Unit 2 and Unit 3 design differences that could potentially impact flooding or spray scenarios were identified during the flooding analysis through reviews of drawings, and the performance of walkdowns. No risk significant differences were identified. The internal flood analysis took into consideration the impact from flood sources in the opposite unit, the potential for floods causing a dual unit shutdown, and minimal design differences between units.

<u>PRA-RAI-4 Part (a)</u> Response Summary

Unit differences were considered during the internal flood analysis. These differences were appropriately addressed, meeting ASME SR requirements. IFPRA Internal flood scenario contributors to CDF, initiating event frequencies and total IFPRA CDF (5.1% of CDF for both units) are similar for both units. No substantive differences exist between the IFPRA DNPS Unit 2 and Unit 3 modeling and results that are judged to affect the conclusions of the PRA.

<u>PRA-RAI-4</u> PRA Differences (part b)

b. Provide an explanation of how the results from the DNPS, Unit 2 PRA model are sufficient for assessing dual unit risk for this LAR.

<u>EGC Response (part b):</u>

The results from the DNPS, Unit 2 PRA model are sufficient for assessing dual unit risk for this LAR for the following reasons:

As previously noted in response to Part (a) the overall CDF for Unit 2 and Unit 3 are the same to 3 significant digits.

The DNPS Self-Assessment and Internal Flood peer review found multi-unit ASME/ANS Standard Requirements to be Met for the DNPS PRA model. These requirements include the following:

- IE-A10 CAT I-III: For multi-unit sites with shared systems, INCLUDE multi-unit site initiators (e.g., multi-unit LOOP events or total loss of service water that may impact the model.
- IE-B5 CAT I-III: For multi-unit sites with shared systems, DO NOT SUBSUME multi-unit initiating events if they impact mitigation capability.

- See also the Internal Flood multi-unit SRs found in response to Part (a), above.

The unit differences are not significant, resulting in very similar risk profiles. The Unit 2 PRA model is the lead model and its system fault trees and event trees are used to create the Unit 3 PRA model. A step in creating the U3 PRA model is to change some fault trees to reflect changes in power dependencies. The quantification of the results of the Unit 3 PRA model show no significant impacts in the risk profile. Examples of configuration differences resulting in modeling changes include AC and DC power dependencies.

These differences have minimal impact on the risk profile of Unit 3 versus Unit 2. The Unit 2 and Unit 3 configuration differences are well understood and documented. Given the insignificant change to risk profile, it remains appropriate to apply Unit 2 ILRT risk results to address the Unit 3 surveillance interval extension.

***PRA-RAI-4 Part (a)(b)* Response Summary**

The Unit 2 Full Power Internal Events (FPIE) PRA modeling, including the Internal Flood PRA modeling, takes into account the impact of Unit 3 initiating events and component unavailability and system status. This includes flooding events in Unit 3 that may impact Unit 2. The Unit 2 and Unit 3 configuration differences are minimal and understood. The 2013 Unit 2 and Unit 3 FPIE PRA overall CDF and LERF results and risk profiles have insignificant differences. Therefore, using the Unit 2 PRA model is appropriate for assessing dual unit risk for this LAR.

***PRA-RAI-5* Accident Sequence Category Assumptions**

In Table 4.2-5, "Accident Sequence Category Descriptions from the License Renewal Severe Accident Management Alternatives Evaluation," of Attachment 3 (page 4-20), the licensee indicated that the "time of end of release" for consequence category L2-10 (intact containment) is 36 hours. This assumption is critical to the determination of the estimated change in population dose for the extended ILRT interval application.

The NRC staff requests the licensee to explain the likelihood of further releases after 36 hours, and if likely, provide an updated assessment of the releases against the population dose criteria for internal as well as external events.

EGC Response:

The ILRT risk assessment utilizes the DNPS-specific dose consequence results developed in support of the License Renewal Severe Accident Management Alternatives (SAMA) Evaluation which was reviewed by the NRC as part of the License Renewal approval process. The intact containment consequence category L2-10 was based on Modular Accident Analysis Program (MAAP) Case DR 0043, with the calculation run to a time of 36 hours. MAAP Case DR 0043 models a loss of RPV makeup with the RPV at low pressure. Upon vessel breach core spray and suppression pool cooling are initiated. Drywell sprays are not initiated. The core spray injection cools the core debris sufficiently to preclude the ex-vessel debris from reaching the Mark I containment liner and failing containment via a shell liner melt-through (i.e., water injection post RPV breach is required to prevent containment failure.) The presence of water

upon the core debris reduces the release of fission products to the environment through postulated containment specification leakage pathways as the debris is cooled. This mitigating effect is demonstrated through the MAAP calculated fission product release (i.e., release fraction) to the environment as a function of time as shown in the figures below for MAAP Case DR 0043 for three notable fission product groups (i.e., CsI, CsOH, and noble gases) .

Figure 5-1 shows the cumulative release fraction released to the environment via technical specification leakage for CsI, where iodine is recognized to be a significant contributor to early dose. The release plot plateaus by approximately 12 hours. Figure 5-2 shows similar results for CsOH, where Cs is recognized to be a significant contributor to long term dose. The plot shows the rate of release decreasing substantially after 8 hours. Figure 5-3 shows the results for noble gases which are generally unaffected by water addition and therefore continue to increase in release in effectively a linear fashion. Noble gases do not significantly contribute to long term doses, which are the major constituents of the SAMA population dose results (i.e., individuals living long term on marginally contaminated land), since noble gases are non-depositing.

To confirm the potential impact of noble gas release variations on population dose, several sensitivity cases were conducted using DNPS specific MACCS2 files substantially similar to those used in the final SAMA analysis. (The MACCS2 files used in the final SAMA analysis were not readily available to support the current sensitivity analysis. The files used for this sensitivity analysis were the initial (draft) MACCS2 files developed to support the SAMA.). The results of the sensitivity cases are presented and discussed in Table RAI.5-1.

TABLE RAI.5-1: MACCS2 SENSITIVITY CASES FOR NOBLE GAS RELEASE

MACCS2 CASE	50-MILE POPULATION DOSE (PERSON-REM)	COMMENTS
DRESDcheck.out (Sensitivity Base case)	1.98E+03	Initial run to form base case. No changes made to inputs. Results compare closely to the final SAMA MACCS2 value of 2.08E+03 (i.e., within 5%)
DRESDNGx2.out (NG release is doubled; plume duration not changed)	2.00E+03	Increasing the noble gas release by 100% (a factor of two) increases the total population dose by approximately 1%.
DRESDPL4.out (NG release extended for additional 24 hours at previous release rate)	1.99E+03	Continuing the noble gas release an additional 24 hours (i.e., added another 24 hour duration plume) increases total population dose by approximately 0.5%.
DRESDNoNG.out (NG release set to zero)	1.96E+03	Noble gas releases eliminated, decreasing the total population dose by about 1%. Noble gas releases are thus a small contributor to the 50-mile population dose.

The sensitivity cases confirm that the contribution of noble gases to the 50-mile population dose is very small, and increasing the noble gas release through either a larger release or a longer release has a negligible impact.

It is additionally noted that if other accident mitigation measures were included in the modeling, such as drywell sprays, the release to the environment would be expected to be further reduced. An alternate DNPS intact containment MAAP case (DR050508a) is similar to the SAMA MAAP Case DR 0043, except drywell sprays are credited instead of RPV core spray. The calculated release to the environment is substantially lower for DR050508a (crediting DW sprays) as shown in Table RAI.5-2.

TABLE RAI.5-2: CONTAINMENT INTACT MAAP CASE RELEASE FRACTION COMPARISON

MAAP CASE	WATER CREDIT	CSI RELEASE FRACTION TO ENVIRONMENT	CSOH RELEASE FRACTION TO ENVIRONMENT
DR 0043 (SAMA Case used in ILRT risk assessment)	RPV Core Spray @ RPV breach	6.3E-06 (@ 36 hrs)	2.2E-06 (@ 36 hrs)
DR050508a (Alternate Intact Containment case)	Containment Spray @ RPV breach	1.2E-07 (@ 38 hrs)	1.1E-06 (@ 38 hrs)

Based on the above, the likelihood of further release after 36 hours is judged to be very small for fission products of interest, and thus there is a minimal impact upon the total population dose and therefore a negligible impact upon the ILRT risk assessment. Additionally, the MAAP case selected for the SAMA analysis and used in the ILRT risk assessment is conservative for intact containment sequences where Drywell sprays would be available.

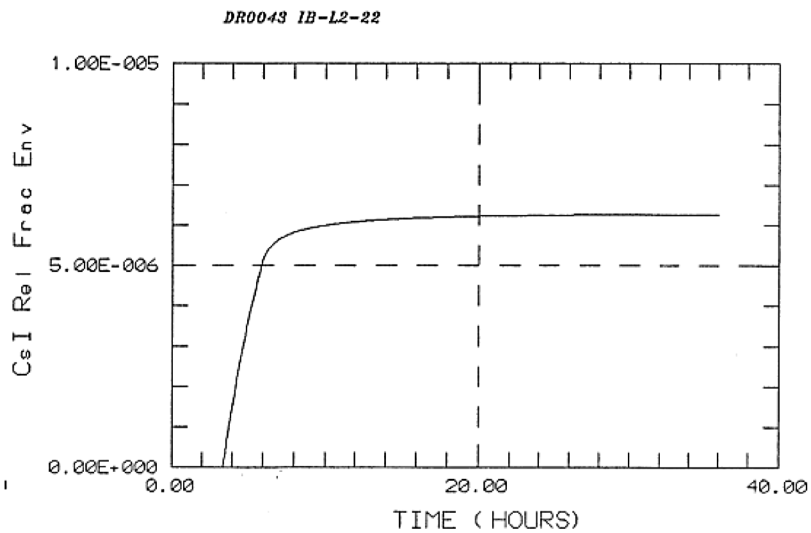

FIGURE 5-1: CSI RELEASE FRACTION TO THE ENVIRONMENT

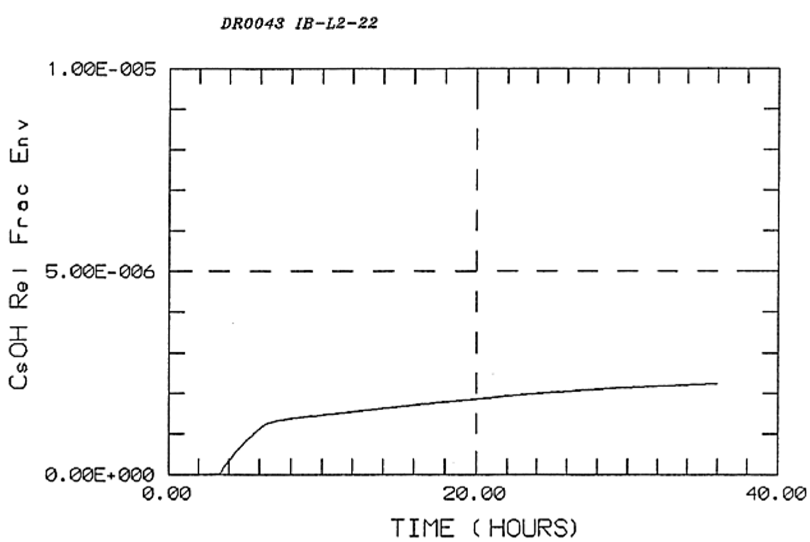

FIGURE 5-2: CSOH RELEASE FRACTION TO THE ENVIRONMENT

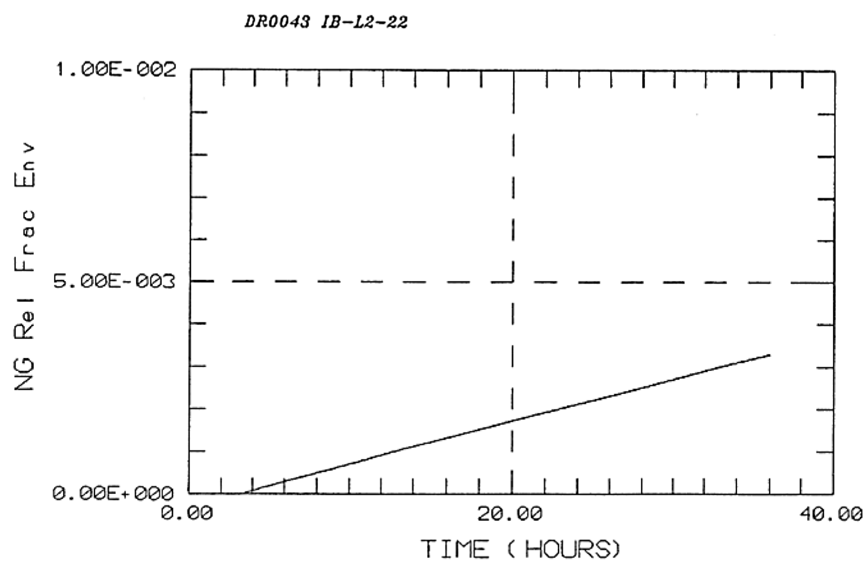

FIGURE 5-3: NOBLE GAS RELEASE FRACTION TO THE ENVIRONMENT

<u>PRA-RAI-6</u> Containment Accident Pressure (CAP) Credit (part a)

In Section 3.3.3, "DNPS 10 CFR 50, Appendix J, Option B Licensing History," Table 3.4.1-1, of Attachment 1 (page 13 of 78), the licensee states that it relies upon containment over-pressure for emergency core cooling system (ECCS) performance and for pressure effects on net positive suction head (NPSH) for the low pressure [coolant] injection (LPCI) and [core] spray (CS) pumps, and that details are provided in Section 3.2, of Attachment 1 of the LAR. In Section 5.8, "Containment Overpressure Impacts on CDF," of Attachment 3 (pages 5-36 and 5-37), provides an estimate of the increase in internal events core damage frequency (CDF) by applying the Event Class 3b probability of large preexisting containment leakage at 15 years to the containment isolation failure probability in the PRA model.

It is not clear to the NRC staff how increasing the containment isolation failure probability accounts for the increase in CDF for accident scenarios that credit NPSH for success, including the following example accident scenarios identified in Electric Power Research Institute (EPRI) Report No. 1009325, Revision 2, "Risk Impact Assessment of Extended Integrated Leak Rate Testing Intervals":

- *Loss of coolant accident (LOCA) scenarios where the initial containment pressurization helps to satisfy the NPSH requirements for early injection, and*

- *Total loss of containment heat removal scenarios where gradual containment pressurization helps to satisfy the NPSH requirements for long-term use of an injection system from a source inside containment.*

The NRC staff requests that the licensee:

a. Identify all accident sequences, and provide a brief description of each, that rely on NPSH to prevent core damage. Describe how loss of containment accident pressure (CAP)-related NPSH is included in the PRA model and evaluated for CDF and LERF for these sequences. If only increasing containment isolation failure probability is assessed, provide justification.

<u>EGC Response (part a):</u>

Identification of All Accident Sequences that rely on NPSH

All accident sequences that include failure of Core Spray (CS) and Low Pressure Coolant Injection (LPCI) pump trains taking suction from inside containment are modeled as relying on net positive suction head (NPSH) and are modeled in the event trees described in Table RAI.6-2. Loss of NPSH is a contributor to sequences as noted in Column 2 "Loss of SPC/SDC and Containment Integrity Leads to Loss of CS and LPCI." Column 3 provides a brief description of the Loss of NPSH sequence, which is a loss of injection scenario.

MAAP runs described in response to PRA-RAI-6 Part (c) were completed and show that when suppression pool cooling (SPC) is available the reduction of NPSH associated with a pre-existing leak is not sufficient to fail the LPCI or CS pumps (i.e., sufficient NPSH remains to support continued pump operation).

PRA Modeling of Loss of Containment Accident Pressure (CAP) Related NPSH

In a scenario where decay heat removal using SPC or Shutdown Cooling (SDC) is not available, the torus will heat up and threaten a loss of NPSH to the CS and LPCI pumps. In this scenario, a breach of containment will fail both CS and LPCI injection. As modeled in the Dresden DR213A PRA model, a pre-existing leak is a breach of containment. Therefore, the present PRA model appropriately increases the core damage contribution of those sequences that credit CS and LPCI injection. CS and LPCI pump suction from outside the containment is not credited in the PRA model.

Section 5.8 CONTAINMENT OVERPRESSURE IMPACTS ON CDF, of the ILRT LAR submittal Attachment 3, utilizes guidance found in Electric Power Research Institute (EPRI) 1009325 Revision 2-A. This guidance states, "As a first order estimate of the impact, it can be assumed that the EPRI Class 3b contribution would lead to loss of containment overpressure and the systems that require this contribution to NPSH should be made unavailable when such an isolation failure exists". For the DNPS PRA model, the current containment isolation failure logic probability was increased by the Class 3b frequency at 15 years (i.e., 0.0023 * 5.0 = 0.0115) to estimate a bounding increase in CDF. The increase in frequency was applied to basic event 2CNHU-PREINITH-- "Pre-existing Containment Failure" basic event. The probability of 2CNHU-PREINITH-- increased from 2.70E-3 to 1.42E-02. With this increase applied, the CDF increased by 5.45E-08/year.

This calculated CDF increase, used in Section 5.8, overstates the impact of loss of NPSH, since loss of NPSH is already included in the base PRA model. The basic event 2CNHU-PREINITH-- is used in the Level 1 PRA for the purpose of failing CS and LPCI when SPC or SDC is not available. The impact of loss of NPSH in the base model due to a pre-existing containment failure can be ascertained by setting the probability of the event to 0. The results are shown in Table RAI.6-1 below:

TABLE RAI.6-1: IMPACT OF LOSS OF NPSH DUE TO A PRE-EXISTING LEAK

2CNHU-PREINITH-- PROBABILITY	CORE DAMAGE FREQUENCY (/YR.)	COMMENTS
2.70E-03	3.218E-06	The pre-initiator probability is based on earlier data. It is reasonably close to the base probability of 2.3E-03 using updated EPRI data.
0	3.206E-06	CDF obtained by setting 2CNHU-PREINITH-- to "False" in the DR213A baseline cutset file.
Delta Risk	1.2E-08	

The delta risk caused by loss of NPSH, is 1.2E-08 /year. The dominant risk contributor is a Class ID sequence (86.5%), i.e., a loss of injection with the RPV at low pressure. This Class ID sequence results in a LERF sequence in the Dresden PRA model because event 2CNHU-PREINITH-- is modeled as a containment isolation failure in the Level 2 DNPS PRA; this leads to a "Large/Early" release. The Level 2 PRA modeling is conservative, as MAAP cases show CS and LPCI loss of NPSH sequences are not early and are not high magnitude

releases. This modeling conservatism is discussed in additional detail in the PRA-RAI-7 response.

PRA-RAI-6 Containment Accident Pressure (CAP) Credit (part b)

b. Confirm that the PRA model used for the CAP risk assessment included all initiating events or conditions which correspond to the two EPRI guidance general event categories.

EGC Response (part b):

PRA modeled accident scenarios are incorporated through the use of event trees. Each event tree was reviewed to determine if CS and LPCI NPSH concerns were appropriately addressed. In all cases where CS and LPCI injection from the torus was credited, injection from this source inside containment was defeated early in the sequence under the following circumstance:

- Loss of SPC and SDC coupled with a containment isolation failure.

EPRI identifies the following accident scenarios for consideration:

- Loss of coolant accident (LOCA) scenarios where the initial containment pressurization helps to satisfy the NPSH requirements for early injection, and
- Total loss of containment heat removal scenarios where gradual containment pressurization helps to satisfy the NPSH requirements for long-term use of an injection system from a source inside containment.

The first EPRI scenario listed above is evaluated using bounding MAAP calculations. The MAAP calculations are described in detail in response to PRA-RAI-6 Part (c). The evaluation found that core damage would not occur for the following scenario:

- SPC available, and
- Pre-existing leak (200 La to demonstrate margin), and
- CS or LPCI available with injection from inside containment (torus).

The assessment supports the current PRA modeling.

The second EPRI scenario is a total loss of containment heat removal scenario where gradual containment pressurization helps to satisfy the NPSH requirements for long-term use of an injection system from a source inside containment. The Dresden PRA model does not credit CS or LPCI injection with a total loss of containment heat removal and a pre-existing leak (i.e., there is no credit for gradual containment pressurization).

TABLE RAI.6-2: ACCIDENT SEQUENCES REVIEW

EVENT TREES AND INITIATING EVENT GROUPINGS	**LOSS OF SPC/SDC AND CONTAINMENT INTEGRITY LEADS TO LOSS OF CS AND LPCI**	**SEQUENCE FAILURES ASSOCIATED WITH RPV WATER INVENTORY FUNCTION**
TRANSIENT EVENT TREES		
Turbine Trip With Bypass	Yes	GTR Event Tree(1)
Loss of Feedwater	Yes	GTR Event Tree(1)
MSIV Closure	Yes	GTR Event Tree(1)
Loss of Condenser Vacuum	Yes	GTR Event Tree(1)
Manual Shutdown	Yes	GTR Event Tree(1)
Inadvertent Open Relief Valve (IORV) or Stuck Open Relief Valve (SORV)	Yes	IORV/SORV Event Tree
Single Unit Loss of Offsite Power (LOOP)	Yes	LOOP Event Tree
Dual Unit Loss of Offsite Power (DLOOP)	Yes	DLOOP Event Tree
LOOP/DLOOP and SORV	Yes	SORV with LOOP an SORV with DLOOP Event Trees
LOCA EVENT TREES		
Small Break Loss of Coolant Accident (LOCA)	Yes	
Medium Break Loss of Coolant Accident (LOCA)	Yes	
Large Break Loss of Coolant Accident (LOCA)	Yes	
LOCAs Outside Containment, and Interfacing Systems LOCA (ISLOCA)	Yes	Initiator results in containment bypass, therefore, a pre-existing leak would not impact CDF
Excessive LOCA	Yes	LPCI not credited.
ATWS EVENT TREES		
ATWS with Main Condenser Available	N/A	CS/LPCI not credited in this event tree.
ATWS with Main Condenser Unavailable	Yes	
ATWS with Main Condenser Unavailable and Stuck Open Relief Valve	Yes	

TABLE RAI.6-2: ACCIDENT SEQUENCES REVIEW

EVENT TREES AND INITIATING EVENT GROUPINGS	LOSS OF SPC/SDC AND CONTAINMENT INTEGRITY LEADS TO LOSS OF CS AND LPCI	SEQUENCE FAILURES ASSOCIATED WITH RPV WATER INVENTORY FUNCTION
LOOP/DLOOP ATWS	Yes	
SPECIAL INITIATOR EVENT TREES		
• Loss of Two 125V DC Buses • Loss of SW/Loss of Station Cooling • Loss of Instrument Air • Loss of Nitrogen • Reactor Water Level Reference Leg Failure • Loss of Drywell cooling • Loss of TBCCW • Loss of Bus 21 (31) • Loss of Bus 22 (32) • Loss of Bus 23 (33) • Loss of Bus 24 (34) • Loss of Bus 28 (38) • Loss of MCC 28-2 (38-2) • Torus or Connected Pipe Rupture • Torus Suction Pipe Rupture in a Corner Room • DGCW, FPS, or CCSW Pipe Rupture in Torus Area • Service Water Rupture in Reactor Building on Elevation 545' or above • Circulating Water Break in Condenser Pit (T.B.) • Service Water Rupture in Turbine Building on Elevation 595' or above • Pipe Spray of Buses 23 or 24 • Service Water, DGCW, FPS, or CCSW Rupture in a Corner Room • FPS, Fuel Pool, or CCST Pipe Rupture in Reactor Building on Elevation 545' or above • FPS or DGCW Ruptures in Turbine Building • CCSW Pipe Rupture in Turbine Building	Yes	GTR Event Tree[1]

Table RAI.6-2 Comments:

(1) Special Initiators are treated in the General Transient (GTR) Event Tree unless consequential events occur resulting in transfer to one of the event trees listed below.
- SORV Event Tree
- Large LOCA Event Tree
- Failure to Scram Event Tree
- Failure to Scram Event Tree with an SORV

***PRA-RAI-6* Containment Accident Pressure (CAP) Credit (part c)**

c. Describe thermal hydraulic analyses performed in support of loss of CAP evaluation, and how they are credited in the PRA if applicable.

EGC Response (part c):

Dresden Specific MAAP Calculations

Table RAI.6-3 below presents the results of two MAAP calculations performed to confirm that sufficient NPSH exists to support continued pump operation for the LPCI and CS systems if containment heat removal via SPC is available. These MAAP calculations used bounding assumptions and inputs with regards to the ILRT risk assessment methodology, as follows:

- Both MAAP cases assume a design basis large break loss of coolant accident (LOCA) (i.e., 28" diameter recirculation line break), which has the greatest potential to increase the temperature of the suppression pool and lead to a challenge due to loss of NPSH.
- A single LPCI pump train and heat exchanger is operating in SPC mode.
- Pre-existing leakage is assumed to be 200 La (i.e., rather than 100 La specified in the ILRT methodology) to demonstrate margin with respect to postulated leakage.
- Both LPCI and CS pump RPV make-up were evaluated.

These MAAP calculations demonstrate that with a single train of containment heat removal operating the suppression pool temperature remains well below 212°F, (a temperature that can challenge NPSH requirements). The MAAP results show that even for a 200 La containment leakage rate, the suppression pool temperature remains below 165°F, and the pumps have sufficient NPSH to remain operational.

These MAAP calculations were reviewed for inputs and key assumptions that could potentially be non-conservative and impact the loss of NPSH assessment.

Key assumptions and inputs examined include:

1. LPCI SPC heat removal start time – For each MAAP calculation SPC is initiated 10 minutes after the suppression pool temperature rises above 95°F. This takes into account time for alignment and is consistent with the current DNPS emergency

operating procedures (EOPs), which require operators to hold suppression pool temperature below 95°F.

2. Initial suppression pool water temperature – The initial water temperature is conservatively set at 95°F, the highest temperature allowed by the EOPs without needing suppression pool cooling.
3. Initial pool water inventory – The initial suppression pool water level is set at a best-estimate height of 14.8 feet which is consistent with typical plant operation.
4. Containment Cooling Service Water (CCSW) temperature - The CCSW ultimate heat sink water temperature is assumed to be 95°F based on the DNPS Updated Final Safety Analysis Report (UFSAR) and is a conservative value.
5. Reactor Power level – Reactor power level is set at 100%, the highest rated core power level, for all calculations.
6. Large LOCA MAAP Modeling issues – MAAP is known to have some modeling issues (e.g., potential for reverse flow not modeled) that introduce uncertainty for large LOCA scenarios. These uncertainties only are applicable to results in the early portion of the run (i.e., approximately first three minutes) prior to core recovery. These deficiencies do not impact the ILRT MAAP calculation results, since the peak suppression pool temperature is reached hours into each run, which is well beyond the calculation timeframe during which the uncertainty is greatest.
7. Pump flow rates – Pump flow rates are based on system analysis data for flow capabilities of the respective pumps.
8. MAAP NPSH calculation – MAAP evaluates, and will fail, injection sources based on NPSH requirements. The MAAP NPSH calculation includes control volume pressures and vertical heights (i.e., static head) but ignores flow losses (e.g., pipe friction and line losses). Ignoring pipe flow losses is non-conservative, but these losses are very small contributors in comparison to the factors included in the MAAP NPSH calculation. In the DNPS MAAP calculations, the NPSH margin available is large compared to the potential impact of line losses.

The above assumptions and inputs are considered to be either best estimate or conservative, except for the non-conservative modeling aspect associated with pipe flow losses, which is judged to be a negligible contributor. Based on the above, the MAAP calculations are believed to present a best estimate result for evaluating the potential loss of NPSH. As noted previously, the MAAP case assumptions of postulating a large LOCA and applying a containment leakage of 200 L_a provide margin in the MAAP calculation results. It is additionally noted that industry testing and analysis indicate that emergency core cooling system ECCS pumps used in boiling water reactor (BWR) 3/4 plants are capable of adequate short term (i.e., ~24 hr) operation well below the manufacturer's recommended design NPSH (e.g., 65% of the specified NPSH limit for Browns Ferry as documented in NUREG/CR-2973, "Reactor Safety Study: An Assessment of Accident Risks in U.S. Commercial. Nuclear Power Plants"). Therefore, additional margin exists beyond that reflected in the MAAP calculations.

TABLE RAI.6-3: MAAP RESULTS[(1)]

LLOCA CASE	CONTAINMENT LEAK SIZE EQUIVALENCE	INJECTION LOST AT TIME	TIME TO CORE DAMAGE	PEAK SUPPRESSION POOL TEMP.	INITIAL CONDITIONS
DR-ILRT-LLOCA-CS-200TS-	200 La	N/A	N/A	163°F	1 CS pumps, 1 LPCI pump in SPC mode
DR-ILRT-LLOCA-LPCI-200T	200 La	N/A	N/A	164°F	1 LPCI pump in LPCI mode (1 loop), 1 LPCI pump in SPC mode[(2)]

Notes to Table RAI.6-3:

(1) All MAAP simulation durations are 24 hours.
(2) The DNPS LPCI system consists of two loops with 2 LPCI pump trains and 1 heat exchanger in each loop. DNPS identifies the system as LPCI and does not use the term ‘RHR" in identifying trains and components.

Conclusions

MAAP calculations performed in support of the DNPS PRA demonstrate that if SPC containment heat removal is available, the suppression pool water temperature stays well below 212 °F in the long term for a large LOCA and reduction in NPSH for LPCI and CS injection is not a concern (i.e., the LPCI and CS pumps continue to operate). Table RAI.6-3 above presents the results from the two bounding MAAP sensitivity cases performed in support of the ILRT analysis.

The PRA model appropriately does not fail CS and LPCI injection with suction from the torus, given a pre-existing leak and either SPC or SDC in operation.

***PRA-RAI-6* Containment Accident Pressure (CAP) Credit (part d)**

d. If containment cooling is credited for NPSH concerns, explain how this credit is taken in the PRA model given the assumption of a pre-existing leak.

EGC Response (part d):

Credit is given to containment cooling to address NPSH concerns in sequences where there is a pre-existing leak. Should SPC and SDC not be available, a pre-existing leak would cause loss of CS and LPCI injection. The PRA model includes an "AND" gate, ECCS-NPSH ‘CS AND LPCI PUMPS FAIL DUE TO LOSS OF NPSH’, that is "true" when SPC, SDC and a pre-existing leak are "true". This gate fails CS and LPCI injection early in a sequence. If other credited RPV inventory control systems are unavailable, core damage and a large early release occur. The impacted sequences are described in the response to RAI-6 Part (a).

***PRA-RAI-6* Containment Accident Pressure (CAP) Credit (part e)**

e. It appears that CAP-related risk does not include external events contribution. Include CAP-related risk in the evaluation of external events risk, and describe the approach.

EGC Response (part e):

CAP related risk reported in Section 5.8 of Attachment 3 of the DNPS ILRT LAR submittal did not include external events contributions. CAP related risk in the evaluation of external events risk, and the approach for considering CAP related risk, are addressed in the response to Part (f) below.

***PRA-RAI-6* Containment Accident Pressure (CAP) Credit (part f)**

f. Evaluate the CAP-related LERF contribution and describe the method used. If a PRA model is used, provide justification for credits which result in a reduction of LERF risk. If a PRA model was not used, confirm that the EPRI Report No. 1009325 allowed first order approximation method of assuming that the change in CDF equals the change in LERF. If a different method had been taken, describe and provide justification.

EGC Response (part f):

A PRA model was used to determine an increase in CAP-related CDF. This increase in CAP-related CDF is modeled as containment isolation failure and is therefore assumed to result in LERF. The value presented in Section 5.8 of Attachment 3 of the LAR submittal is 5.45E-08/year. This increase in NPSH related LERF is further refined.

The pre-existing leak is included in the Level 1 FPIE PRA model as basic event 2CNHU-PREINITH--. The basic event fails CS and LPCI due to NPSH concerns when SPC and SDC functions are not available. The baseline CDF contribution from a pre-existing leak (i.e., represented by basic event 2CNHU-PREINITH--) is 1.2E-08/year as identified in Table RAI.6-1. (See EGC Response, part a.)

An adjustment is made to the CDF contribution as the pre-existing leak probability used in the 2013A PRA is 2.7E-03 (based on earlier EPRI ILRT data), rather than 2.3E-3 (today's value). The number of ILRT demands increased from 182 to 217. Using the Jeffreys non-informative pre-existing leak probability based on 217 demands results in a pre-existing leak probability of 2.3E-03. Therefore, the baseline CDF NPSH related contribution is adjusted to account for the lower leak probability. The CDF contribution becomes:
1.2E-08/year * (2.3E-03 ÷ 2.7E-03) = 1.0E-08/year.

Based on the ILRT surveillance frequency change, the increase in CDF (LERF) in going from 3 in 10 year frequency to 1 in 15 year frequency is:
[(5 * base CDF contribution) – base CDF contribution] = 4 X 1.0E-08/year = 4.0E-08/year.

This 4.0E-08/year LERF increase is used in determining the increase in LERF and total LERF in the response to PRA-RAI-8.

PRA-RAI-7 (Formerly PRA-RAI-9) Overall LERF Risk Assessment (part a)

In Table 5.7-5, "DNPS External Events Contributor Summary," of Attachment 3 (page 5-53), the following data is depicted in the last two rows:

	CDF(1/yr)		***LERF(1/yr)***
Total for External Events	*4.98E-5*		*8.58E-6*
Internal Events	*3.22E-6*		*7.31E-7*

Summing external and internal frequencies results in a total CDF of 5.30E-5/year and a total LERF of 9.21E-6/year. The total LERF is close to the RG 1.174, "An Approach for Using Probabilistic Risk Assessment in Risk-Informed Decisions on Plant-Specific Changes to the Licensing Basis," Revision 2, (ADAMS Accession No. ML100910006), guideline of 1E-5/year.

The NRC staff requests that the licensee provide additional information regarding the assumptions and uncertainties in its estimate:

a. Assess quantitatively or qualitatively the key assumptions or sources of uncertainty on the total LERF to include the following sources and provide a discussion of the assessment results:

1. *Fire contribution,*
2. *CAP-related contribution,*
3. *Seismic contribution,*
4. *Steel liner corrosion contribution, and*
5. *Class II decay heat removal contribution.*

EGC Response (part a):

The response to PRA RAI 8 calculates a total LERF of 7.08E-06/year rather than 9.21E-06/year, thereby providing more quantitative margin to the total LERF acceptance criterion of 1.0E-05/year. For RAI 7, the sources identified above have been assessed qualitatively with regards to impact of key assumptions or sources of uncertainty on total LERF. A discussion related to each source follows:

1. *Fire Contribution*

   Assumption 1: The Individual Plant Examination External Events (IPEEE) fire risk profile adequately represents the current as-built, as operated plant.

   Discussion: As further detailed in response to 7(b), a number of plant changes and changes to operating practices made since the completion of the IPEEE in 1999 result in a reduction in risk. Therefore, using a Fire PRA representing the as-built, as-operated plant in 1999 is conservative.

Assumption 2: The IPEEE Fire CDF accurately represents fire risk.

Discussion: The IPEEE fire risk provided a CDF, however, the intent of the analysis was to identify the most risk significant fire areas in the plant using a screening process and by calculating conservative core damage frequencies for fire scenarios.

Section 5.7.1 of Attachment 3 of the ILRT LAR submittal describes attributes of a Fire PRA that are considered conservative or bounding. These include:

- The IPEEE used fire initiating event data from the pre-2000 timeframe. In comparison, the latest NRC fire events database indicates trends toward both lower frequency and less severe fires.
- Fire protection measures such as sprinklers, CO_2, and fire brigades may be given minimal (conservative) credit in their ability to limit the spread of a fire.
- Limited or no credit is given to Balance of Plant (BOP) systems due to lack of cable data or detailed circuit analysis. (The FPIE PRA model results show that credit for BOP systems is important in CDF mitigation.)
- The number of fire scenarios may be minimized to reduce analytic burden (e.g., multiple fire ignition sources grouped into a single fire scenario).
- Fire modeling presents bounding approaches regarding fire propagation and the fire's immediate effects (e.g., fire impacted cables and components are failed at time t = 0; all cables in an impacted tray are failed).

In the DNPS ILRT LAR, the Unit 3 Fire IPEEE CDF risk was used since it was larger than the Unit 2 CDF (i.e., by approximately a factor of 1.8). Therefore, in view of fire risk associated with Unit 2, there is additional conservatism.

As further discussed in response to PRA-RAI-7, site initiatives to reduce fire risk have been taken since the completion of the IPEEE PRA. These initiatives include addressing fire risk in the 10 CFR 50.65(a)(4) program, site training focused on fire risk, and in-plant labeling of "High" Fire Risk areas.

As in any PRA, there is uncertainty. The following would lead to under estimating fire risk.

- The IPEEE fire risk methodology screened out low risk areas. Table 4-10 of the DNPS IPEEE Submittal Report (Rev. 1, February 2000) lists the fire compartments that were quantitatively screened from the IPEEE Fire CDF based on a bounding calculated CDF at the compartment level. For Unit 3, the total CDF of quantitatively screened compartments was 5.25E-07/year. (The Unit 2 total CDF of quantitatively screened areas was lower). For Unit 3, this represents the potential for an additional 1.7% CDF, above the 3.08E-05/year value used in the ILRT risk assessment. Given the conservative bounding nature of the quantitative screening approach used, the additional CDF contribution is very small and of negligible risk importance to the conclusions of the ILRT risk assessment.

- The Multiple Spurious Operation (MSO) Process and the related MSO Generic List (Nuclear Energy Institute (NEI) 00-01) were not explicitly addressed in the IPEEE as the MSO industry guidelines did not exist at the time the IPEEE was prepared. However, the IPEEE development also did not explicitly limit or otherwise constrain the fire risk assessment to consider only single spurious operations. Given the inherent modeling of events in the PRA, combinations of spurious events are therefore included. As an example, the IPEEE did include consideration of the fire induced spurious opening of one and more SRVs as well as generation of a spurious ADS signal. A quantitative estimate of the uncertainty associated with not including all MSOs per the latest industry guidance is not available.

Based on the key assumptions above, use of the IPEEE Fire PRA is considered a reasonable approach.

2. CAP-related Contribution

The CAP related key assumptions and uncertainty are discussed in response to RAI-7 Part (c), which is in a subsequent section.

3. Seismic Contribution

Uncertainty exists for the DNPS seismic CDF as a reduced-scope seismic margins analysis (SMA) was provided as part of the IPEEE program. There is currently limited detailed information regarding plant seismic capacity (i.e., the ability of the plant's structures, systems and components to successfully withstand an earthquake) beyond the required design-basis level. To address this uncertainty, a conservative approach to selecting a seismic CDF was used in the DNPS ILRT risk assessment. Additional detail follows:

The DNPS ILRT risk assessment used a seismic CDF estimate from the NRC GI-199 report (i.e., Reference 28 of Attachment 1 of the LAR submittal). To address uncertainty, the ILRT risk assessment used as input, the highest seismic CDF (1.9E-05/year) provided for DNPS. The GI-199 report included estimated seismic CDFs based on three sets of mean seismic hazard curves that were generated at various times and by various organizations as follows:

1. Electric Power Research Institute, 1989
2. Lawrence Livermore National Laboratory, 1994
3. NRC based on U. S. Geological Survey, 2008

Eight seismic CDFs were estimated for each set of seismic hazard curves, as documented in Tables D-1, D-2, and D-3 and described in Section 3.1 "Seismic Core-Damage Frequency Estimates of GI-199." Thus GI-199 produced a total of 24 seismic CDF estimates applicable to both DNPS, Unit 2 and Unit 3.

From the 24 seismic CDF estimates from the NRC's GI-199 report, DNPS selected the highest CDF value, namely, the "weakest link" value of 1.90E-05/year. Alternative seismic CDF estimates include the following:

- NRC GI-199 CDF values of:
  - Table D-1, seismic CDF estimates using 2008 USGS seismic hazard curves
    - Simple average – 1.1E-05/year.
    - IPEEE weighted average – 9.7E-06/year.
    - Lowest estimate (1 Hz spectral frequency) – 2.4E-06/year.
  - Table D-2, Seismic CDF estimates using 1989 EPRI seismic hazard curves
    - Simple average – 1.1E-06/year.
    - IPEEE weighted average – 9.5E-07/year.
    - Lowest estimate (1 Hz spectral frequency) – 1.7E-07/year.
  - Table D-3: Seismic CDF estimates using 1994 LLNL hazard curves
    - Simple average – 5.3E-06/year.
    - IPEEE weighted average – 4.9E-06/year.
    - Lowest estimate (1 Hz spectral frequency) – 1.9E-06/year.

Review of the range of seismic CDF estimates available for DNPS (i.e., 1.7E-07/year. to 1.9E-05/year. from GI-199) shows that there is significant uncertainty based on the selection of the seismic hazard curve and spectral frequencies. Use of the highest published value in the ILRT risk submittal bounds the lowest published value by two orders of magnitude and is judged to adequately address seismic risk uncertainty.

A key assumption impacting Seismic ILRT risk impact is that the Seismic Core Damage Frequency risk profile is similar to the FPIE risk profile. This assumption supports using the FPIE internal events LERF multiplier to estimate the seismic ILRT LERF. Use of the bounding seismic CDF value provides margin in applying the LERF multiplier approach.

4. Steel Liner Corrosion Contribution

Steel corrosion could increase flaw likelihood. The uncertainty is addressed in Section 6.1 of the ILRT submittal, Attachment 3. As documented in Section 6.1, sensitivity cases were developed to gain an understanding of the sensitivity of the results to the key parameters in the corrosion risk analysis. The time for the flaw likelihood to double was adjusted from every five years to every two years and then every ten years. The failure probabilities for the wall and basemat were increased and decreased by an order of magnitude. The total detection failure likelihood was adjusted from 10% to 15% and 5%. The results are presented in Table 6.1-1 of Section 6.1. In every case, the impact from including the corrosion effects is minimal. Even the upper bound estimates with conservative assumptions for all of the key corrosion parameters yield an increase in LERF due to corrosion of only 1.27E-07/year compared to a base case increase due to corrosion of 4.4E-09/year. The bounding analysis is extremely unlikely and

represents a "small" change in LERF in accordance with RG 1.174, "An Approach for Using Probabilistic Risk Assessment in Risk Informed Decisions on Plant-Specific Changes to the Licensing Basis." The base case parameters are already judged to be conservative. The upper bound includes setting each of the key parameters at an even more conservative level than the base case.

The steel liner contribution is a factor of the pre-existing leakage probability. For Class 3b, consistent with latest available EPRI data, a Jeffreys non-informative prior distribution is assumed for no "large" failures in 217 tests (i.e., 0.5/(217+1) = 0.0023). This methodology is considered conservative. Table D-2 of EPRI Guidance document TR-1018243, dated October 2008, Appendix D, "Expert Elicitation Results and Analysis," provides a Jeffreys non-informed prior probability based on 400 tests performed world-wide. This sampling results in a pre-existing leakage probability of 1.25E-03. This indicates the pre-existing leakage probability of 2.3E-03 is conservative.

A large pre-existing leak (i.e., ≥ $100L_a$) would likely be detected by operations staff, since the drywell is inerted during operations. A large pre-existing leak would result in a loss of nitrogen inerting and in-leakage resulting in an increase in oxygen inside containment. The Drywell O_2 Content Hi Alarm actions are addressed for Unit 2 under DAN 923-5A A-4 Revision 10. Probable causes listed in the DAN procedure are 1) incomplete mixing in the Drywell or Torus and 2) Air in-leakage. Operator actions include "Attempt to identify AND isolate the source of air in-leakage into the Containment." Technical Specification 3.6.3.1 Primary Containment Oxygen Concentration requires restoration of O_2 to less than 4% within 24 hours. If not met, the reactor thermal power must be reduced to less than 15%. Therefore, a large pre-existing leak existing for a significant period of time is unlikely.

Steel liner corrosion contribution uncertainty exists as demonstrated by sensitivity cases discussed above. However, the likelihood of an existing leak is conservative based on the methodology used and data beyond the U.S. data collected as input to the Jeffreys non-informed prior calculation. The likelihood of a pre-existing leak going undetected during plant operations is low given that O_2 levels are continuously monitored. Should O_2 levels go above TS limits for 24 hours, the reactor must be brought to low power levels, significantly reducing the likelihood of a LERF event caused by a pre-existing leak.

5. Class II decay heat removal contribution

Class II decay heat removal scenarios are included as contributors to FPIE LERF. The FPIE PRA model assumes a General Emergency (GE) would not be declared in 5% of the Class II sequences. Given the significant time to declare a GE while containment pressure is increasing, the 5% estimate is conservative. In addition, a 100 La leak would relieve pressure and delay failure of containment due to overpressure. MAAP cases without a leak demonstrate that containment failure would occur approximately 23 hours from the start of the Class II event. The DNPS Level 2 analysis for Class II sequences determined that there is more than 24 hours available between the declaration of a GE and a CsI release exceeding 10% (large release threshold). The evacuation time is conservatively estimated to be 6.6 hours. Therefore, there is significant margin with release timing such that Class II sequences will not result in a large, early release. Therefore, the 5% estimate of a large, early release is conservative.

PRA-RAI-7 ***(Formerly PRA-RAI-9) Overall LERF Risk Assessment (part b)***

b. With regard to fire-related risk, in Section 5.7.1, of Attachment 3, explains that the DNPS Individual Plant Examination of External Events (IPEEE) fire analysis was used in the risk assessment of the extended ILRT interval because a fire PRA model was not available. Given that the IPEEE analysis was completed in March 2000, provide a discussion of the impact on the risk assessment results of the current plant configuration and operating experience including a discussion on plant changes that have been made since the IPEEE that would significantly increase/decrease fire risk, including for screened fire areas.

EGC Response (part b):

A review of plant changes made since the completion of the IPEEE was performed by reviewing FPIE PRA model changes made since that time. Based on their potential for significantly impacting the FPRA, plant changes included in the following model updates were reviewed:

- *2005 PRA Model Update*
  No plant modifications identified as key contributors to changes in risk.

- *2009 (DR209A and DR209A) PRA Model Update*
  A risk significant plant modification to install a flood barrier around the 4 kilovolt (kV) critical switchgear in the Reactor Building has been incorporated into the model. – The modification impacted the Internal Flood PRA modeling only. No impact to the Fire PRA risk profile.

- 2013 (DR213A and DR313A) PRA Model Update
  No plant modifications were identified as key contributors to changes in risk.

Other – Updating Requirements Evaluation (URE)

As noted in the ILRT submittal, a PRA updating requirements evaluation URE is created for all issues that could potentially impact the PRA models. UREs are typically initiated by the PRRA model owner, and/or the site risk management engineer, to evaluate plant design changes, and procedure changes that could impact the FPIE and Fire PRA models. The URE database has been maintained since February 1997. The IPEEE Fire PRA CDF used in the DNPS ILRT submittal was reported in the DNPS, Units 2 and 3, IPEEE submittal report Revision 1, dated May 25, 1999. The UREs generated between January 1, 1998, and the present were reviewed to identify changes to the as-built, as-operated plant (e.g., Design Changes, Calculations and Procedure Changes).

There were a number of changes that were made to improve plant equipment reliability. The following examples are provided:

- Instrument Air Compressors replaced and upgraded
- Service Water Pumps replaced and upgraded
- Replacement of Auxiliary Transformers with automatic load tap changing capability
- Replacement of 250 VDC Battery Chargers with capability to plug in a portable battery charger
- Added redundant Shutdown Cooling System temperature isolation logic

Although these changes were not necessarily made to explicitly address fire risk, the improvements in reliability generally reduce the risk due to internal events and fires.

There were a number of procedure changes to enhance operator performance including the incorporation of "Hard Cards", which were added to operating procedures. These Hard Cards are used in the Control Room to give operators key direction, rather than relying on memory or needing to find specific sections in the governing procedures.

The Hard Cards improve the ability of the operators to respond to accident scenarios, including those for which information and mitigation capability may be impacted by a fire. In addition, site procedures have been added to address Fire Risk under the 10 CFR 50.65(a)(4) program. Fire risk insights are now used to identify those areas considered "high-risk" for fire based on plant configurations. Actions are taken to mitigate this risk, including protecting equipment, assuring key equipment is available and increasing operator walk-throughs.

Design changes that provide a risk reduction include the following:

- B.5.b Portable Equipment
- FLEX Equipment
- Addition of a Hardened Containment Vent System (HCVS)

For accident sequences where there is initial success of the isolation condenser (IC) or reactor pressure vessel (RPV) injection, the B.5.b portable equipment and FLEX equipment significantly augment plant capabilities for accident mitigation. Similarly, for Class II accident sequences, incorporation of the HCVS provides additional containment heat removal capability.

Operating Experience

A review was performed in 2014 to identify plant-specific fire events that were judged to be "potentially challenging". The review identified four fires in a span of 13 years (2001—2013). The review findings support the conclusion that DNPS is not an industry outlier from the perspective of fire risk. The data supported use of the IPEEE CDF results as a bounding approach to ILRT risk impact. Additional details follow.

The DNPS fire event reports for the period of January 1, 2001 through December 31, 2013 were reviewed to determine the number, type, plant status, impacts, location, and detection/suppression of "potentially challenging" fires occurring at DNPS. The compiled generic data includes fires that are potentially challenging and therefore any update must use these same evaluation criteria. The review methods and classification criteria are in accordance with Appendix C of NUREG/CR-6850. The fire event review revealed four (4) potentially challenging fire events.

The four events are:

- Bus 31 Fan Motor Shorted Out (3/6/2005): An electrical fault in the cabinet which caused a fire to start in the fan motor assembly.
- Undetermined Fire in Unit 1 Turbine Building Maintenance Shop (9/30/2012): Event classified as a plant wide transient because of the lack of available details of the fire. The fire may have started at a light switch panel and spread to transient items in the room.
- Light Fire Results in Cable Damage (7/10/2013): A light fire resulted in melted material falling into an exposed cable tray and damaging cable insulation. Active intervention extinguished the fire.
- Fire in Bus 21 Cooling Fan (10/21/2008): Smoke and a small flame were observed emanating from a breaker cooling fan.

In summary, these plant specific fire events do not indicate DNPS is an outlier with respect to plant fires.

***RAI 7(b)* Response Conclusion**

The IPEEE analysis was based on data prior to 2000. Reviews of plant changes (i.e., design changes and procedures) since 2000 and fires between 2001 and 2013 were performed. The reviews found no changes or fire challenges that significantly increase fire risk. In fact, given plant equipment reliability modifications, incorporation of Hard Cards for Control Room operator use, the availability of FLEX and Hardened Containment Vent System (HCVS) equipment, and improvements in the 10 CFR 50.65(a)(4) "Maintenance Rule Program," the overall impact of plant configuration and operating experience could significantly decrease fire risk as compared to the IPEEE fire risk.

***PRA-RAI-7* (Formerly PRA-RAI-9) Overall LERF Risk Assessment (part c)**

c. With regard to CAP-related risk, identify any key assumptions or sources of uncertainty for the loss of CAP evaluation, given the assumed pre-existing leak, and discuss the assessment of them.

EGC Response (part c):

Key assumptions or sources of uncertainty for the loss of CAP evaluation are the following:

1. No credit is given for recovery of the Isolation Condenser (i.e., for heat removal and/or inventory control) during the event. This assumption is conservative.
2. No credit is given for alignment of CS and LPCI to the Condensate Storage Tank (CST), a source outside of containment. Procedures are in place to align CS and LPCI to the CST. Aligning the CS and LPCI pumps to the CST would significantly reduce CDF for single unit events. During dual unit events, such as a dual unit loss of offsite power (DLOOP), water inventory may be insufficient to extend CS and LPCI injection beyond 24 hours. However, additional time would be available for an "early" declaration of a General Emergency or recovery of other injection sources. Not crediting CS and LPCI alignment to the CST is conservative.
3. Loss of CS and LPCI from a loss of NPSH due to a pre-existing leak and loss of SPC and SDC is assumed to lead to a large early release (i.e., with other inventory make-up sources unavailable). Credit for an "early" declaration of a General Emergency based on loss of NPSH concerns is not provided. A timeline provided by MAAP Case DR-ILRT-GTR-CS 200TS shows that loss of CS injection (most limiting case) would occur at 4.4 hours and RPV breach at 16.3 hours. Prior to RPV breach, safety relief valves are venting radionuclides to the torus. Scrubbing of the radionuclides in the torus (suppression pool) would limit off-site dose. A large release is assumed at time of RPV breach time. This assumption is conservative.
4. The seismic risk profile is assumed to be similar to the FPIE risk profile. Therefore, the seismic NPSH contribution is estimated using the ratio of seismic CDF to FPIE CDF multiplied by the FPIE NPSH CDF contribution. Given that the highest available seismic CDF value was used, this assumption is conservative.

***PRA-RAI-7* (Formerly PRA-RAI-9) Overall LERF Risk Assessment (part d)**

d. Identify and discuss any other sources of uncertainty which could be important for assessing the total LERF, and include them in the evaluation.

EGC Response (part d):

Additional sources of uncertainty are identified as follows:

1. One source of uncertainty is the ability to detect leakage. This uncertainty was addressed with an assumption that a leakage path allowing ≤ 100 La will go undetected until the next

ILRT. This assumption is conservative given that O_2 levels would increase inside of containment and the increase in oxygen is alarmed in the Control Room. Early detection would result in initiation of correction actions including a possible shutdown, thus precluding a containment condition that could result in a large release via a pre-existing leak pathway.

2. Another source of uncertainty is whether or not a "large" "early" release would occur in a loss of NPSH scenario. As discussed in response to part (c) above, this uncertainty is addressed with the CAP related assumption that loss of low pressure injection (CS and LPCI) due to a pre-existing leak and loss of SPC and SDC leads to a large early release. This is a conservative assumption. Loss of inventory make-up would result in a delayed "non-Large" release. RPV breach occurs approximately 11 hours after a loss of RPV injection due to loss of NPSH. There would be adequate time for evacuation following a declaration of a General Emergency.

***PRA-RAI-7* (Formerly PRA-RAI-9) Overall LERF Risk Assessment (part e)**

e. Assess the cumulative effect of these considerations and provide justification for why the total LERF is consistent with the RG 1.174 guidance when LERF-related assumptions or uncertainties are considered.

EGC Response (part e):

The cumulative impact of the considerations identified above is judged to result in a conservative calculated ILRT LERF. Additionally, the total LERF impact is reduced significantly by an application specific model change (i.e., reclassification of key LERF sequence contributors from large to medium releases) as described in the RAI 8 response. This change to LERF classifications provides additional margin. Key LERF-related assumptions discussed earlier are found to be conservative. These assumptions include the following:

- No credit for Fukushima FLEX strategy or use of the HCVS. Note, the HCVS is in addition to the existing Torus Drywell Vent (TDV). The TDV has instrument air (IA) system dependencies. The HCVS would survive loss of the IA system. This is an important feature in fire modeling. FLEX strategy and HCVS are available to operators, but were not credited in the 2013 FPIE PRA model used to support the ILRT risk assessment.
- The IPEEE Fire PRA CDF is based on conservative modeling techniques.
- The seismic PRA CDF estimate used was the largest available DNPS specific value, and therefore was a conservative input when estimating LERF.
- A pre-existing leak size of ≤ 100 La is likely to be detected during operations.
- The CST suction source for CS and LPCI pumps is not credited in the PRA in the loss of CS and LPCI injection due to loss of NPSH. Crediting this suction source would reduce the ILRT calculated CDF and LERF.
- Loss of CS and LPCI injection due to loss of NPSH and leading to core damage is assumed to result in a "large" "early" release. This is conservative, given that the release is scrubbed between loss of inventory make-up and RPV breach. A significant release would

not occur for many hours, allowing time for evacuation resulting in a "non-early" release. Related MAAP cases are found in Table RAI.7-1 below.

- The steel liner assumptions regarding impact to pre-existing leakage probability have uncertainty that may impact LERF. However, the uncertainty calculations performed for the ILRT LAR submittal demonstrate that the impact would be small.
- Some Class II scenarios are assumed to contribute to LERF. This is conservative given the significant time that exists between containment failure and a large release.

TABLE RAI.7-1: MAAP RESULTS

CASE	DESCRIPTION	TAF	ED (# VALVES)	CD	VESSEL BREACH	CONT. FAILURE	CONT. SPRAY	NOBLE GAS RELEASE FRACTION	CSI RELEASE FRACTION @ 25HR(1)	COMMENTS
DR-ILRT-GTR-CS-100TS	MSIV Close No IC Core Spray 5 SRVs at TAF 100x TS Leakage	26 min.	26 min. (5 SRVs)	9.02 hr.	16.6 hr.	16.7 hr. DW Shell Failure 7 min after VB	N/A	1.0	8.13E-3	Lose CS at 4.98 hr. due to NPSH Pool temp. at 4.98 hr. = 199°F
DR-ILRT-GTR-CS-200TS	MSIV Close No IC Core Spray 5 SRVs at TAF 200x TS Leakage	26 min.	26 min. (5 SRVs)	8.56 hr.	16.3 hr.	16.4 hr. DW Shell Failure 7 min after VB	N/A	1.0	7.36E-3	Lose CS at 4.42 hr. due to NPSH Pool temp. at 4.42 hr. = 193°F
DR-ILRT-GTR-LPCI-100TS	MSIV Close No IC LPCI 5 SRVs at TAF 100x TS Leakage	26 min.	26 min. (5 SRVs)	8.66 hr.	16.1 hr.	16.2 hr. DW Shell Failure 7 min after VB	N/A	1.0	2.61E-3	Lose LPCI at 4.73 hr. due to NPSH Pool temp. at 4.73 hr. = 197°F
DR-ILRT-GTR-LPCI-200TS	MSIV Close No IC LPCI 5 SRVs at TAF 200x TS Leakage	26 min.	26 min. (5 SRVs)	8.51 hr.	16.6 hr.	16.7 hr. DW Shell Failure 7 min after VB	N/A	1.0	9.08E-3	Lose LPCI at 4.76 hr. due to NPSH Pool temp. at 4.76 hr. = 196°F

Note to Table RAI.7-1:

(1) CsI release fraction at 25 hours represents the release magnitude after population evacuation would be completed. Per the Level 2 PRA evaluation, evacuation will be completed in approximately 6.6 hours. Conservatively assuming that a General Emergency declaration is not made until the time of containment failure following vessel breach (rather than upon the time of core damage due to NPSH indicative of containment leakage beyond technical specifications), evacuation would be completed by approximately 23.3 hours (i.e., 16.7 hours + 6.6 hours).

PRA-RAI-8 (Formerly PRA-RAI-10) Total Increase in CDF and LERF

The NRC staff requests the licensee to provide the total increase in CDF and LERF for internal and external events, including the CAP related risk increase.

EGC Response:

The total increase in CDF and LERF for internal and external events, including CAP related risk increase is calculated using the CAP contribution calculated in the response to PRA-RAI-6. Tables from Section 5.7 of Attachment 3 of the DNPS ILRT LAR submittal are updated. An application specific PRA model is used to support the recalculation of the total increase in CDF and LERF. The application specific PRA model consisted of changing two LERF sequences from Large/Early (L/E) to Medium/Early (M/E) release based on Dresden-specific MAAP calculations (i.e., DR050505 and DR0505a). Both sequences involve loss of RPV injection with the RPV depressurized. Case DR050505 shows that CsI release fraction would not exceed 10% until after 22 hours. An evacuation would be completed in less than seven (7) hours, providing significant margin to a L/E release based on timing (i.e., approximately 15 hours). Also, insights from the NRC State-of-the-Art Reactor Consequence Analyses (SOARCA) project (i.e., see Section 5.1 of NUREG/CR 7110, "State-of-the-Art Reactor Consequence Analyses," Vol. 1, Rev. 1, May 2013 for the Peach Bottom plant) show a high probability of a core damage induced SORV. Case DR050500a shows that a SORV leads to more fission product retention in the suppression pool, limiting the release magnitude to the environment to medium (i.e., CsI release fraction of 1.34E-02 at 38 hours). Therefore, a M/E release is assessed for this MAAP case.

Changes to the model ID-041 and IBE-041 sequences impact the LERF results. CDF results are not impacted. The DR213A PRA LERF cutsets include sequence tags. The revised LERF is obtained by setting the ID-041 and IBE-041 sequence tags to false. With this change DR213A LERF is changed from the baseline value of 7.31E-07/year to 4.77E-07 /year.

This analysis impacts results found in Section 5.7 of Attachment 3 of the ILRT LAR submittal, beginning with Section 5.7.4. That section is repeated below, but with updated values identified via use of italicized text. Section numbering is maintained in this RAI response for ease of reference to the ILRT LAR submittal.

5.7.4 External Events Impact Summary

In summary, the combination of the fire and seismic CDF values described above results in an external events bounding risk estimate of 4.98E-05/year, as shown in Table 5.7-5. Seismic and Fire LERF values derived from CDF values in sections 5.7.2 and 5.7.3 and as shown in Table 5.7-5 below sum to a LERF value of *5.60E-06/year*, which is *11.7* times higher than the internal events LERF.

Table 5.7-5 summarizes the estimated bounding external events CDF contribution for DNPS.

TABLE 5.7-5 (RAI 8 CHANGES ITALIZED): DNPS EXTERNAL EVENTS CONTRIBUTOR SUMMARY

EXTERNAL EVENT INITIATOR GROUP	CDF (1/YR)	LERF (1/YR)
Seismic	1.90E-05	*2.81E-06*(1)
Internal Fire	3.08E-05	*2.78E-06*(2)
High Winds	Screened	Screened
Other Hazards	Screened	Screened
Total For External Events (for initiators with CDF/LERF available)	**4.98E-05**	***5.60E-06***
Internal Events CDF - LERF (for comparison)	3.22E-06	*4.77E-07*

Notes to Table 5.7-5:

(1) The IPEEE seismic analysis did not report LERF. Seismic LERF is assumed to be the ratio of seismic CDF to FPIE CDF multiplied by FPIE LERF (i.e., (1.90E-05/year ÷ 3.22E-06/year * *4.77E-07/year*)). See Section 5.7.2 for further discussion.

(2) The IPEEE Fire PRA model did not report LERF. Fire LERF is assumed to be the ratio of Fire CDF to FPIE CDF (without Class II non-LERF contribution) multiplied by FPIE LERF (i.e., (1.81E-05/year ÷ 3.10E-06/year * *4.77E-07/year*)). See Section 5.7.1 for further discussion.

As noted earlier, the 3b contribution is approximately proportional to CDF. An increase in CDF would likely lead to higher 3b frequency and assumed LERF. The Fire CDF contributors were adjusted to remove Class II scenarios where an "early" declaration of a General Emergency was achieved. The sequence contribution for the Seismic CDF is unknown and no adjustments were made. To determine a suitable multiplier of external event CDF to internal event CDF, a multiplier is developed for each external event group (i.e., fire and seismic) and then added together to address both contributors, as shown in Table 5.7-6. For fire contribution, the adjusted CDF (i.e., Class II scenarios removed) ratio of fire and FPIE is multiplied by the portion of FPIE CDF that the fire contribution can act upon (i.e., ratio of adjusted FPIE CDF and unadjusted FPIE CDF). For seismic, the ratio of unadjusted CDF (i.e., seismic and FPIE) is used.

TABLE 5.7-6: DNPS EXTERNAL EVENTS TO INTERNAL EVENTS CDF COMPARISON

EXTERNAL EVENT INITIATOR GROUP	CDF (1/YR)	EXTERNAL EVENT ADJUSTED CDF (1/YR)	FPIE INITIATOR GROUP	FPIE CDF (1/YR)	INITIAL MULTIPLIER	APPLICABLE MULTIPLIER PORTION(2)
Fire	3.08E-05	1.81E-05(1)	FPIE (without Class II sequences)	3.10E-06(1)	5.8	5.6
Seismic	1.90E-05	NA	FPIE (unadjusted)	3.22E-06	5.9	5.9
External Event to FPIE CDF Multipler						**11.5**

Notes to Table 5.7-6:

(1) Class II sequence CDF contributions that "cannot result in LERF" are not included.

(2) The initial fire multiplier is reduced by a factor of 0.963 (i.e., 3.10E-06/year / 3.22E-06/year) because the initial fire multiplier is only applicable to a portion of the unadjusted FPIE CDF (3.22E-06/year). The initial seismic multiplier is based on the unadjusted FPIE CDF and therefore no further reduction factor is applied.

5.7.5 External Events Impact on ILRT Extension Assessment

The EPRI Category 3b frequency for the 3-per-10 year, 1-per-10 year, and 1-per-15 year ILRT intervals are shown in Table 5.6-1 as 7.59E-09/year, 2.61E-08/year, and 4.07E-08/year, respectively. Using an external events LERF multiplier of 11.5 (i.e., multiplier from Table 5.7-6) for DNPS, the change in the LERF risk measure due to extending the ILRT from 3-per-10 years to 1-per-15 years, including both internal and external hazards risk, is estimated as shown in Table 5.7-7 below.

TABLE 5.7-7: DNPS 3B (LERF/YEAR) AS A FUNCTION OF ILRT FREQUENCY FOR INTERNAL AND EXTERNAL EVENTS (INCLUDING AGE ADJUSTED STEEL LINER CORROSION LIKELIHOOD)

	3B FREQUENCY (3-PER-10 YR ILRT)	**3B FREQUENCY (1-PER-10 YEAR ILRT)**	**3B FREQUENCY (1-PER-15 YEAR ILRT)**	**LERF INCREASE[1]**
Internal Events Contribution	7.59E-09	2.61E-08	4.07E-08	3.31E-08
External Events Contribution (Internal Events x 11.5)	8.74E-08	3.00E-07	4.69E-07	3.81E-07
NPSH Internal Events (RAI 6f)	-	-	-	*4.0E-08*
NPSH External Events (Internal Events x 11.5) (RAI 6f)	-	-		*4.60E-07*
NPSH Combined (Internal + External	-	-	-	*5.00E-07*
Combined (Internal + External)[2] + NPSH	-	-	-	***9.14E-07***

Note to Table 5.7-7:

[1] Associated with the change from the baseline 3-per-10 year frequency to the proposed 1-per-15 year frequency.

[2] Calculations were performed using more than 3 significant figures. Therefore, figures may differ in the 3rd digit if one adds the figures shown above.

The other metrics for the ILRT extension risk assessment can be similarly derived using the multiplier approach. The results between the 3-in-10 year interval and the 15 year interval compared to the acceptance criteria are shown in Table 5.7-8. As can be seen, the impact from including the external events contributors would not change the conclusion of the risk assessment. That is, the acceptance criteria are all met such that the estimated risk increase associated with permanently extending the ILRT surveillance interval to 15 years has been demonstrated to be small. Note that a bounding analysis for the total LERF contribution follows Table 5.7-8 to demonstrate that the total LERF value for DNPS is less than 1.0E-05/year consistent with the requirements for a "Small Change" in risk of the RG 1.174 acceptance guidelines.

TABLE 5.7-8: COMPARISON TO ACCEPTANCE CRITERIA INCLUDING EXTERNAL EVENTS CONTRIBUTION FOR DNPS

CONTRIBUTOR	**ΔLERF /YR**	**ΔPERSON-REM /YR**	**ΔCCFP[(1)]**
Internal Events	3.31E-08	4.26E-02 (0.27%)	1.03%
External Events	3.81E-07	4.91E-01 (0.27%)[(2)]	1.03%
NPSH (Int + Ext)	*5.00E-07*	*4.93E-01[(3)] (0.25%)[(4)]*	*1.03%*
Total	***9.14E-07***	***1.03 (<0.52%)[(5)]***	***1.03%***
Acceptance Criteria	**<1.0E-6/yr ("small")**	**Least restrictive of <1.0 person-rem/yr or <1.0%**	**<1.5%**

Note to Table 5.7-5:

(1) The probability of leakage due to the ILRT extension is assumed to be the same for both Internal and External events. Therefore, the percentage change for CCFP remains constant.

(2) Calculated as the FPIE value times the external events multiplier of 11.5 developed in Table 5.7-6.

(3) Dose from NPSH assumes EPRI Class 3b dose: 9.86E+05 per-rem * 5.00E-07 ΔLERF /year = 4.93E-01 Δper-rem /year.

(4) The NPSH percentage of Δperson-rem to baseline person-rem is calculated based on FPIE values and assumes the same for external events:
FPIE ΔLERF /year of 4.0E-08/year * 3b dose of 9.86E+05 / 15.785 per-rem/year of Table 5.6-1 for 3 in 10 Year ILRT Frequency.

(5) The total percentage of Δperson-rem to baseline person-rem is calculated as 1.03 per-rem/year total increase divided by the internal events person-rem for 3 in 10 year frequency (15.785 per-rem/year.) plus the external events person-rem for 3 in 10 year. frequency, which is
11.5 * FPIE person-rem (11.5 * 15.785 person-rem/year = 181.5 person-rem/year).
% person-rem/year = 1.03 person-rem/year ÷ (15.785 + 181.5) person-rem/year
% person-rem/year = 0.52%

The 9.14E-07/year increase in LERF due to the combined internal and external events from extending the ILRT frequency from 3-per-10 years to 1-per-15 years falls within Region II between 1.0E-7 to 1.0E-6 per reactor year ("small" change in risk) of the RG 1.174 acceptance guidelines. Per RG 1.174, when the calculated increase in LERF due to the proposed plant change is in the "small" change range, the risk assessment must also reasonably show that the total LERF is less than 1.0E-5/year. Similar bounding assumptions regarding the external event contributions that were made above are used for the total LERF estimate.

From Table 4.2-2 of Attachment 3 of the LAR submittal, the LERF (High Early) due to postulated internal event accidents is 7.31E-07/year for DNPS. However, an application specific model change reduces LERF to 4.77E-07/year. As shown in Table 5.7-5, the LERF estimate for the Fire PRA model is 2.78E-06 /year and the total LERF estimate for the Seismic PRA model is 2.81E-06 /year. The total LERF values for DNPS are shown in Table 5.7-9 below.

TABLE 5.7-9: IMPACT OF 15-YEAR ILRT EXTENSION ON LERF FOR DNPS

LERF CONTRIBUTOR	**TOTAL LERF (1/YR)**
Internal Events LERF	*4.77E-07*
Fire LERF	*2.78E-06*
Seismic LERF	*2.81E-06*
Internal Events LERF due to ILRT (at 15 years) (1)	*4.07E-08*
External Events LERF due to ILRT (at 15 years) (1)	4.69E-07 [Internal Events LERF due to ILRT * 11.5]
NPSH LERF due to the ILRT (2)	*5.00E-07*
Total	***7.08E-06/yr***
Acceptance Criterion	**<1.0E-05/yr**

<u>Notes to Table 5.7-9</u>:

(1) Including age adjusted steel liner corrosion likelihood as reported in Table 5.7-7.

(2) NPSH LERF is equal to the NPSH ΔLERF. NPSH LERF from a pre-existing leak is included in the Internal Events CDF of *4.77E-07/year*.

Response to PRA-RAI-8 Summary

The total increase in CDF and LERF for internal and external events, including the CAP related risk increase is below the RG 1.174 thresholds. As can be seen, the estimated total LERF for DNPS is *7.08E-06/year*. This value is less than the RG 1.174 requirement to demonstrate that the total LERF due to internal and external events is less than 1.0E-05/year. The ILRT delta LERF of *9.14E-07*, which includes the CAP related a risk increase, remains below the RG 1.174 threshold of 1.0E-06/year.

The total increase in LERF and total LERF estimates are considered "bounding" as a significant contributor is the CAP related risk increase. The CAP related LERF contribution is judged to be overstated as adequate time is available for population evacuation in the CAP related loss of low pressure injection scenario, as discussed in the response to RAI 7. Therefore, this contribution can be assessed as a non-early release.

The calculated LERF and delta LERF values presented in the RAI-8 response provide additional margin to the RG 1.200 thresholds when compared to the margins that existed in the original ILRT submittal calculations. It should be noted that this additional margin is with the inclusion of the CAP related loss of NPSH non-LERF sequences. Also significant is that more margin would exist in delta LERF if the CAP related contribution was not included. The delta LERF would be reduced from approximately 9E-07 to 4E-07/year, providing a margin of 6E-07/year to the RG 1.174 threshold of 1E-06/year.

SBPB-RAI-1

The NRC staff notes that the licensee adopted Option B of 10 CFR 50, Appendix J following the January 11, 1996, issuance of Amendment No. 148 to Facility Operating License No. DPR-19 and Amendment No. 142 to Facility Operating License DPR-25 for DNPS, Units 2 and 3, respectively. Per the guidance of NEI 94-01 Revision 0, Section 10.2.3.2 and subject to the four provisions identified in Regulatory Position "C" of Regulatory Guide 1.163, both DNPS Unit 2 and Unit 3 are currently allowed to extend the test intervals for Type C CIVs up to 60 months.

In Section 10.2.3.2 of NEI 94-01 of both Revision 0 and Revision 3-A reads in part:

> *Test intervals for Type C valves may be increased based upon completion of two consecutive periodic As-found Type C tests where the result of each test is within a licensee's allowable administrative limits.*

In Section 3.5.5, "Type B and Type C Local Leak Rate Testing Program Implementation Review," of Attachment 1 (page 55 of 78), the NRC staff notes that a significant minority (i.e., approximately 45%) of the population of both DNPS, Unit 2 and Unit 3 Type C tested components (CIVs) are not on an extended frequency of 60 months. The ability to extend the Type C test intervals from 60 months to 75 months would be supported by having a large percentage of CIVs already qualifying for 60 months with some margin.

The NRC staff requests that the licensee provide additional information about regarding Sections 3.5.4 and 3.5.5 of the LAR. In particular, do the Type C test results indicate that a large percentage of valves have not successfully made it to a 60 month test interval frequency?

EGC Response:

For Type C tests, there are a total of 93 tests (i.e., multiple or single valve tests) for each unit. Of the 93 tests:

- 50 tests are on a 60-month test interval frequency.
- 18 tests cannot be extended in accordance with RG 1.163, "Performance-Based Containment Leak-Test Program"
- 18 tests cannot be extended because of Inservice Testing (IST) Program requirement for bi-direction testing or position indication.
- Seven (7) tests remain on a 30-month frequency because of maintenance requests to overhaul every refuel outage.

Conclusion: Subtracting the RG 1.163 and IST tests (i.e., 36 total), there are 57 Type C tests for each unit that can be extended to 60-month intervals. Of the 57 tests, seven (7) are not on extended frequency due to plant decision to overhaul the valves every refuel outage. Therefore,

of the Type C tests, 87% of the population eligible for extension for each unit is on an extended test frequency.

SBPB RAI-2

DNPS TS 5.5.12a indicates that the leakage rate testing program will be in accordance with Regulatory Guide 1.163 which endorsed NEI 94-01. Determining the As-found minimum pathway combined Type B and C test totals verifies that the requirement had been met at all times when containment integrity was required. In approving the industry proposal to allow ILRT intervals to routinely extend to a maximum of 15 years, the NRC staff required a change to the NEI 94-01 guidance such that NEI 94-01 Revision 2-A and Revision 3-A state that acceptance criteria for the combined As-found leakage rate for all penetrations subject to Type B or Type C testing "be less than 0.6La" rather than "recommended" to be less than 0.6 L.

In Table 3.5.4-1, "DNPS, Unit 2 Types B and C LLRT Combined As-Found/As-Left Trend Summary," of Attachment 1 (page 54 of 78), the aggregate LLRT values in row: "AF Min Path (scfh)," displays outage totals as "Fraction of La." The NRC staff notes that for refueling outage D2R22 in 2011 and that for refueling outage D2R24 in 2015, the as found minimum pathway that "Fraction of La " values were recorded at 0.928 and 0.701, respectively. In both instances, there is neither an explicit indication in the LAR that this represented a failure to meet the surveillance performance criterion nor that past operability of the primary containment had been evaluated given the As-found Type B and C total exceeding 0.6 La. Both occurrences represent a prima facie entrance into the margin reserved for ensuring the overall primary containment performance criterion of La had not been challenged given that the combined Type B and C leakage did not necessarily account for all containment leakage potential.

The NRC staff requests the licensee provide additional information:

a. For example, corrective actions

EGC Response:

For DNPS, Unit 2 Refuel Outages D2R22, D2R23, and D2R24, the largest contributor to the excessive min-pathway leakages in all three cases was the feedwater check valves. A root cause evaluation was completed in May 2012, and determined the causes of the feed water check valve failures were as follows:

1. The root cause was found to be insufficient assertive engineering. This led to managers accepting and not challenging long-term performance deficiencies.

2. A contributing cause was that the best practices for the design, maintenance, and testing of the valves were not effectively utilized.

Discussion:

Over the last 40 years, since the establishment of the 10 CFR 50, Appendix J, "Primary Reactor Containment Leakage Testing for Water-Cooled Power Reactors," local leak rate testing (LLRT) in 1973, LLRT failures have occurred and been attributed to either the failure of the seat ring-to-body seal or the failure of the disc-to-seat ring seal.

A 2000 root cause evaluation report noted problems with the seat ring-to-body sealing. Failures due to the seat ring-to-body sealing failure mode have decreased (i.e., with the last known failure in 2000) due to the following improvements:

1. Upgraded O-ring material to Kalrez 1050, extending O-ring service life

2. Modified seat ring design to acceptably load O-ring

3. Modified the seat ring hold down clamps to evenly and acceptably retain the seat ring

4. Modified the bolting style on two historically susceptible valves

The failures in the past 10 years have been due to disc-to-seat ring sealing failures and have not significantly decreased. The changes/upgrades implemented in the past 40 years dealt generally with maintenance practices, designs, and testing practices. Quad Cities Nuclear Power Station (QCNPS) also implemented many changes in these areas which resulted in better leakage rate testing results than DNPS. The deviation from the QCNPS check valve practices and performance improvements, by DNPS, is attributed to insufficiently assertive engineering. DNPS check valves experienced an average failure rate of 0.39 while QCNPS check valves experienced an average failure rate of 0.24.

The cause of the Dresden Station Feedwater Check Valve failures to meet Appendix J LLRT limits is the inability to fully seat the valve disc in the seat assembly for testing. During a normal shutdown the traditional testing methodology results in slow closure of the check valves which can result in incomplete seating due to minor wear; however, the design basis function involves rapid closure of the valve disc with high force that fully seats the check valves.

Corrective Actions:

1. Develop valve maintenance and testing best practices. Work with Crane Valves, the DNPS Mechanical Maintenance Department (MMD), and Design Engineering from DNPS, LaSalle County Station, and Exelon Corporate to develop common maintenance and testing practices and internals modifications to optimize check valve performance. Consider possible uniqueness of DNPS, Unit 2 Feed Water Header Outboard Isolation Check Valve, 2-220-62B, and DNPS, Unit 3 Feed Water Header Inboard Isolation Check Valve, 3-220-58A, which have higher failure rates than other valves.

2. Implement valve maintenance best practices.

3. Implement valve testing best practices. Modifications to install additional one (1) inch test taps in the downstream, intermediate, and upstream volumes to aid in water drainage and air pressurization rates during testing.

Implementation of Corrective Actions and Improvements to Leak Rate Test Program (The Administrative Acceptance Criteria for each feedwater check valve is < 45 standard cubic feet per hour (scfh), and the overall 0.6 L_a leakage limit is < 810.507 scfh):

- Unit 3, B Feedwater Loop: Feedwater modification and best practice implementation was completed during D3R23 in November 2014. The post-modification, as-found minimum pathway (min-path) leakage for the Unit 3, B feedwater loop during Refuel Outage D3R24 in November 2016 was 2.77 scfh - satisfactory results.

- Unit 2, B Feedwater Loop: Feedwater modification and best practice implementation was completed during D2R24 in November 2015. The post-modification as-found min-path leakage for the Unit 2, B feedwater loop during D2R25 in November 2017 was 0.84 scfh - satisfactory results.

- Unit 3, A Feedwater Loop: Feedwater outboard valve modification and best practice implementation was completed during D3R24 in November 2016. The post-modification as-found min-path leakage for the Unit 3, A feedwater outboard valve will be determined during D3R25 in November 2018.

- Unit 2, A Feedwater Loop: Feedwater modification and best practice implementation was completed during D2R25 in November 2017. The post-modification leakage for the Unit 2, A feedwater loop will be determined during D2R26 in November 2019.

- Unit 3, A Feedwater Loop: Feedwater inboard valve modification and best practice implementation will be completed during D3R25 in November 2018.

***SBPB RAI-2* (continued)**

b. For improvements to the leak testing program, etc., that justify the requested Unit 2 Type A and Type C test extensions during each of the last three refueling outages, D2R22, D2R23 (in 2013), and D2R24

c. For the combined as-found Type B and C minimum pathway test results that exceeded 0.6L_a by significant margins

<u>EGC Response:</u>

D2R22:

As a result of the As-Found leakage exceeding its administrative limits, the following components were identified for repairs. Included in this list are the As-Found and the As-Left leakage rates, which reflect the leakage rate before and after repairs or adjustments:

Valves Identified	As-Found Leakage	As-Left Leakage
2-0203-1A, "Main Steam Inboard Isolation Air Operated Valve"	49.7 scfh	1A & 2A dry test combined leakage 0.883 scfh
2-0203-2D, "Main Steam Outboard Isolation AO Valve"	83.1 scfh	1D & 2D dry test combined leakage 0.010 scfh
2-0220-58A, "Feed Water Header Inboard Isolation Check Valve"	818 scfh	1.31 scfh
2-0220-62A, "Feed Water Header Outboard Isolation Check Valve"	240 scfh	0.74 scfh
2-0220-58B, "Feed Water Header Inboard Isolation Check Valve"	858.42 scfh	7.70 scfh
2-0220-62B*, "Feed Water Header Outboard Isolation Check Valve"	Undetermined	0.705 scfh
2-1201-1 & 1201-1A, "Reactor Water Cleanup Inlet Isolation Motor-Operated Valve (MOV)"	34 scfh	34 scfh
2-1600-X-108A, "Bellows Drywell Penetration Seal Isolation Condenser X-108A"	58.8 scfh (Test fixture leakage)	0.00 scfh
2-1600-X-116A, "Bellows Drywell Penetration Seal Low Pressure Coolant Injection X-116A"	35.6 scfh (Test fixture leakage)	0.820 scfh

*The other valves in series, 2-0220-58B as-found leakage of 858.42 scfh. The leakage from this valve proved that there was minimal safety significance based on the MIN-PATH.

The components that failed their as-found LLRT during D2R22 and the corrective actions taken to repair them were:

- 2-0203-1A - disassembled/inspected/repaired
- 2-0203-2D - disassembled/inspected/repaired
- 2-0220-58A - replaced valve seat and disk
- 2-0220-62A - replaced valve seat and disk
- 2-0220-58B - replaced valve seat and disk.
- 2-0220-62B - replaced valve seat and disk
- 2-1201-1 - completed DTP 47 Form 47A deferral paperwork
- 2-1601-X-108A - repaired test fixture.
- 2-1601-X-116A - repaired test fixture.

D2R23:

As a result of the As-Found leakage exceeding its administrative limits, the following components were identified for repairs. Included in this list are the As-Found and the As-Left leakage rates, which reflect the leakage rate before and after repairs or adjustments:

Valves Identified	As-Found Leakage	As-Left Leakage
2-0220-58B	Undetermined (1)	16.8 scfh
2-0220-62B	Undetermined (1)	0.85 scfh

(1) This undetermined maximum pathway leakage rate was due to one primary containment pathway, the Feed Water Header Outboard Isolation Check Valves, 2-0220-58B and 2-0220-62B. These test volumes could not hold pressure due to the excess leakage (LER 13-005-00).

The components that failed their as-found LLRT during D2R23 and the corrective actions taken to repair them are:

- 2-0220-58B - replaced valve seat and disk.
- 2-0220-62B - replaced valve seat and disk

D2R24:

As a result of the As-Found leakage exceeding its administrative limits, the following components were identified for repairs. Included in this list are the As-Found and the As-Left leakage rates, which reflect the leakage rate before and after repairs or adjustments:

Valves Identified	As-Found Leakage	As-Left Leakage
2-0203-2A, "Main Steam Outboard Isolation Air Operated Valve"	196 scfh	0.021 scfh
2-0203-2B, "Main Steam Outboard Isolation Air Operated Valve"	53.4 scfh	0.581 scfh
2-0203-1D, "Main Steam Inboard Isolation Air Operated Valve"	80.6 scfh	28.7 scfh
2-0220-58A, "Feed Water Header Inboard Isolation Check Valve"	868.4 scfh	0.431 scfh
2-0220-58B, "Feed Water Header Inboard Isolation Check Valve" (1)	Undetermined	0.431 scfh
2-0220-62B, "Feed Water Header Outboard Isolation Check Valve"	819 scfh	4.46 scfh
2-8526, "Nitrogen Makeup Header Relief Valve"	155 scfh	0.915 scfh

(1) 2-0220-62B, the other valves in series, as-found leakage of 819 scfh, provided that there was minimal safety significance based on the MIN-PATH.

The components that failed their as-found LLRT during D2R24 and the corrective actions taken to repair them are:

- 2-0203-1D - upgraded liner, new plug and stem skim cut
- 2-0203-2A - upgraded liner, new plug and stem skim cut
- 2-0203-2B - installed new stem and plug assembly
- 2-0220-58A - flushed.
- 2-0220-58B - disassembled, inspected and repaired.
- 2-0220-62B - disassembled, inspected and repaired
- 2-8526 - replaced relief valve.