FENOC™
FirstEnergy Nuclear Operating Company

Beaver Valley Power Station
P.O. Box 4
Shippingport, PA 15077

Eric A. Larson
Site Vice President

724-682-5234
Fax: 724-643-8069

November 4, 2014
L-14-343

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

SUBJECT:
Beaver Valley Power Station, Unit Nos. 1 and 2
Docket No. 50-334, License No. DPR-66
Docket No. 50-412, License No. NPF-73
Response to Request for Additional Information Regarding License Amendment Request to Extend Containment Leakage Rate Test Frequency (TAC Nos. MF3985 and MF3986)

By letter dated April 16, 2014 (Accession No. ML14111A291), FirstEnergy Nuclear Operating Company (FENOC), requested a license amendment to the facility operating license for Beaver Valley Power Station, Unit No. 1 (BVPS-1) and Unit No. 2 (BVPS-2). The proposed license amendment would revise Technical Specification 5.5.12, "Containment Leakage Rate Testing Program," Item a, by deleting reference to the BVPS-1 exemption letter dated December 5, 1984 (Accession No. ML003766713), and requiring compliance with Nuclear Energy Institute (NEI) topical report NEI 94-01, Revision 3-A, "Industry Guideline for Implementing Performance-Based Option of 10 CFR Part 50, Appendix J," instead of Regulatory Guide 1.163, "Performance-Based Containment Leak-Test Program," including listed exceptions.

By correspondence dated October 1, 2014 (Accession No. ML14259A448), the Nuclear Regulatory Commission (NRC) staff requested additional information to complete its review. Attachment 1 of this letter presents a response to the NRC staff request.

The FENOC response to the request for additional information includes changes to the Technical Specification wording proposed in the April 16, 2014 FENOC letter. Technical Specification 5.5.12 text marked to show the revised wording is provided in Attachment 2. Attachment 3 presents Technical Specification 5.5.12 with the proposed changes incorporated.

There are no regulatory commitments contained in this submittal. If there are any questions or if additional information is required, please contact Mr. Thomas A. Lentz, Manager - Fleet Licensing, at (330) 315-6810.

I declare under penalty of perjury that the foregoing is true and correct. Executed on November 4, 2014.

Sincerely,

E. A. Larson

Eric A. Larson

Attachments:
1. Response to October 1, 2014 Request for Additional Information
2. Proposed Technical Specification Change (Mark-up)
3. Proposed Technical Specification Change (Re-Typed)

cc: NRC Region I Administrator
NRC Resident Inspector
NRC Project Manager
Director BRP/DEP
Site BRP/DEP Representative

FirstEnergy Nuclear Operating Company (FENOC), submitted a license amendment request for Beaver Valley Power Station (BVPS), Unit Nos. 1 (BVPS-1), and 2 (BVPS-2) in a letter dated April 16, 2014 (Agencywide Documents Access and Management System [ADAMS] Accession No. ML14111A291). The proposed amendment would delete reference to the BVPS-1 exemption letter dated December 5, 1984 (ADAMS Accession No. ML003766713), and extend the Type A reactor containment test interval required by 10 CFR 50, Appendix J, from one test in 10 years to one test in 15 years for both BVPS-1 and BVPS-2. The proposed amendment would also extend the Type C test interval up to 75 months, based on acceptable performance history as defined in NEI 94-01, Revision 3-A (ADAMS Accession No. ML12221A202), "Industry Guideline for Implementing Performance-Based Option of 10 CFR Part 50, Appendix J."

In a letter dated October 1, 2014 (ADAMS Accession No. ML14259A448), Nuclear Regulatory Commission (NRC) staff in the Probabilistic Risk Assessment Licensing Branch (APLA), the Containment and Ventilation Branch (SCVB), and the Mechanical and Civil Engineering Branch (EMCB) requested additional information to complete their review of the license amendment request. The information requested is shown below in bold text and is followed by the FENOC response.

APLA RAI 1:

According to Regulatory Issue Summary 2007-06 (ADAMS Accession No. ML070650428), "Regulatory Guide 1.200 Implementation," the NRC staff expects that licensees fully address all scope elements with Revision 2 of Regulatory Guide (RG) 1.200 [ADAMS Accession No. ML090410014], "An Approach for Determining the Technical Adequacy of Probabilistic Risk Assessment Results for Risk-Informed Activities," by the end of its implementation period (i.e., one year after the issuance of Revision 2 of RG 1.200). Revision 2 of RG 1.200 endorses, with exceptions and clarifications, the combined American Society of Mechanical Engineers (ASME)/American Nuclear Society (ANS) PRA standard (ASME/ANS RA-Sa-2009).

The methodology provided in Electric Power Research Institute (EPRI) Technical Report (TR) 1009325, Revision 2 [ADAMS Accession No. ML072970208], "Risk Impact Assessment of Extended Integrated Leak Rate Testing Intervals," to confirm the risk impact of the Integrated Leak Rate Testing (ILRT) extension on a plant-specific basis relies on use of internal events and the available plant-specific risk analyses for external events. EPRI Report No. 1009325, Revision 2, does not address plant-specific probabilistic risk assessment (PRA) quality. In the safety evaluation report (SER) for the EPRI TR-1009325, Revision 2, dated June 25, 2008 [ADAMS Accession No. ML081140105], the NRC staff, stated, in part, that:

> **[l]icensee requests for a permanent extension of the ILRT surveillance interval to 15 years pursuant to NEI TR 94-01, Revision 2, and EPRI Report No. 1009325, Revision 2, will be treated by NRC staff as risk-informed license amendment requests. Consistent with information provided to industry in Regulatory Issue Summary 2007-06, "Regulatory Guide 1.200 Implementation," . . . the NRC staff will expect the licensee's supporting Level 1/[large early release frequency] LERF PRA to address the technical adequacy requirements of RG 1.200, Revision 1 . . . Any identified deficiencies in addressing this standard shall be assessed further in order to determine any impacts on any proposed decreases to surveillance frequencies. If further revisions to RG 1.200 are issued which endorse additional standards, the NRC staff will evaluate any application referencing NEI TR 94-01, Revision 2, and EPRI Report No. 1009325, Revision 2, to examine if it meets the PRA quality guidance per the RG 1.200 implementation schedule identified by the NRC staff.**

Given that the implementation date of RG 1.200, Revision 2, was April 2010 and the license amendment request (LAR) was submitted in April 2014, identify any gaps between the BVPS PRA models used in this application and RG 1.200, Revision 2 requirements, that are relevant to this submittal and address the technical adequacy requirements of RG 1.200, Revision 2, or explain why addressing the requirements would have no impact on this application.

Response:

As noted in Section 2.3 and 2.4 of LAR Attachments 5 and 6, both the BVPS-1 and BVPS-2 probabilistic risk assessment (PRA) models have resolved all of the applicable findings and observations (F&Os) identified in the 2002 BVPS PRA Peer Review, 2007 BVPS PRA self-assessment, and 2007 BVPS human reliability analysis (HRA) focused peer review, to meet the Capability Category II Supporting Requirements (SR) in ASME RA-Sb-2005, as amplified by RG 1.200, Revision 1. Additionally, all of the applicable F&Os from the 2011 BVPS internal flood PRA focused peer review were resolved to meet the Capability Category II or better in the combined ASME/ANS PRA standard (RA-Sa-2009), along with the NRC clarifications and qualifications provided in RG 1.200, Revision 2.

Since the BVPS internal events PRA models were reviewed against the ASME RA-Sb-2005 PRA standard, as endorsed by RG 1.200, Revision 1, a gap assessment was performed to account for the differences between those supporting requirements and the supporting requirements provided in Part 2 of the ASME/ANS PRA standard, as endorsed in RG 1.200, Revision 2. To address these differences, Section 3.3 of NEI 05-04 "Process for Performing Internal Events PRA Peer Reviews Using the ASME/ANS PRA Standard," Revision 3 was used as guidance to perform this gap assessment for the BVPS internal events PRA models. The following provides the results of this gap assessment.

1. Overview of High Level Requirement and Supporting Requirement Changes (NEI 05-04 Section 3.3)

   In general, the changes to the ASME/ANS PRA standard high level requirements (HLRs) and SRs in the transition from Addendum B (ASME RA-Sb-2005) through Revision 1, Addendum A (ASME/ANS RA-Sa-2009) were minor and include the following:

   - Incorporation into the ASME/ANS PRA standard issues that were identified by the NRC in RG 1.200, Revision 1,
   - renumbering of the ASME/ANS PRA standard HLRs and SRs to remove deleted SRs and SRs ending with a letter (for example, SR QU-A2a); as listed in Appendix F of NEI 05-04, Revision 3,
   - changes in the cross-references updated to the new tables, and
   - corrections of typographical and grammar errors, and changes in wording.

   However, there were a few examples of changes to either the ASME/ANS PRA standard or the RG 1.200, Revision 2 that would require re-evaluation of the PRA against the ASME/ANS PRA standard requirements. These are discussed in the following sections.

2. Supporting Requirements Requiring Re-evaluation (NEI 05-04 Section 3.3.1)

   SRs that require re-evaluation are those SRs that have changed significantly, including those with new issues identified in RG 1.200, Revision 2; these SR are provided in Table 1.

Table 1. Supporting Requirements Requiring Gap Assessment Re-evaluation

ASME/ANS RA-Sa-2009 Supporting Requirement	NEI 05-04, Revision 3, Table 3-2 Comments	RG 1.200, Revision 1 to Revision 2 Gap Assessment Re-evaluation and Capability Category (CC)
HR-D6	RG 1.200, Revision, 2 provides clarification that should be evaluated.	Meets: The BVPS HRA models characterize the uncertainty in the estimates of the human error probabilities (HEPs) consistent with the quantification approach and use mean values in the quantification of the PRA results. Uncertainty cases are also provided using the 5th and 95th percentiles of the HEPs.

Table 1. Supporting Requirements Requiring Gap Assessment Re-evaluation (Continued)

ASME/ANS RA-Sa-2009 Supporting Requirement	NEI 05-04, Revision 3, Table 3-2 Comments	RG 1.200, Revision 1 to Revision 2 Gap Assessment Re-evaluation and Capability Category (CC)
HR-G3	RG 1.200, Revision 2, provided clarification to items (d) and (g) of the SR. Some of the RG 1.200, Revision 1 wording remains, while some additional clarification is provided.	CC II/III: The BVPS HRA models use the EPRI HRA calculator, which includes a discussion of the specific scenario to evaluate; the (d) degree of clarity of the cues/indications in supporting the detection, diagnosis, and decision-making give the plant-specific and scenario-specific context of the event, and (g) complexity of detection, diagnosis and decision-making, and executing the required response.
New DA SR	RG 1.200, Revision 1, included a new SR -- DA-D8. The recommended new SR is included in RG 1.200, Revision 2, as DA-D9 (with the renumbering).	Meets: The BVPS PRA models only take credit for repairing the emergency diesel generators (EDGs) in the electric power recovery (EPR) model. This EPR model uses a convolution methodology to calculate the probability of recovering offsite power or repairing an EDG in time to prevent core damage as a function of the accident sequence in which the SSC failure appears.
QU-A2	Need to ensure QU-A2 evaluates LERF results.	Meets: The BVPS PRA models provide estimates of the individual sequences in a manner consistent with the estimation of core damage frequency (CDF) and LERF to identify significant accident sequences and confirm that the logic is appropriately reflected. These estimates are accomplished by using RISKMAN through event trees with conditional split fractions.

Table 1. Supporting Requirements
Requiring Gap Assessment Re-evaluation (Continued)

ASME/ANS RA-Sa-2009 Supporting Requirement	NEI 05-04, Revision 3, Table 3-2 Comments	RG 1.200, Revision 1 to Revision 2 Gap Assessment Re-evaluation and Capability Category (CC)
QU-A3	Need to ensure QU-A3 evaluates LERF results.	CC II: The BVPS PRA models are quantified using Monte Carlo simulations. RISKMAN enables the mean CDF and LERF to be estimated by correlating event probabilities. When propagating uncertainty distributions, the CDF and LERF are estimated.
QU-B5	RG 1.200, Revision 2, provides clarification that should be evaluated. Need to verify breaking logic loops does not result in undue conservatism.	Meets: Both RG 1.200, Revision 1, Table A-1. "Staff Position on ASME RA-S-2002, ASME RA-Sa-2003, and ASME RA-Sb-2005," and RG 1.200, Revision 2, Table A-2. "Staff Position on ASME/ANS RA-Sa-2009 Part 2, Technical and Peer Review Requirements for At-Power Internal Events" have "No objection" to SR QU-B5. Furthermore, the BVPS PRA model logical loops are broken in a manner that still permits each dependency to be accounted for when quantified using event trees with conditional split fractions.
QU-B6	Need to ensure QU-B6 evaluates LERF results.	Meets: The RISKMAN event tree linking quantification process that is used by the BVPS PRA models account for system successes in addition to system failures in the evaluation of accident sequences to the extent needed for realistic estimation of CDF and LERF. This accounting is accomplished by using numerical quantification of success probability. Since the event trees are linked, all "successes" are transferred between event trees.

Table 1. Supporting Requirements Requiring Gap Assessment Re-evaluation (Continued)

ASME/ANS RA-Sa-2009 Supporting Requirement	NEI 05-04, Revision 3, Table 3-2 Comments	RG 1.200, Revision 1 to Revision 2 Gap Assessment Re-evaluation and Capability Category (CC)
QU-E3	Need to ensure QU-E3 evaluates LERF results.	CC II: The BVPS PRA models take into account the "state of knowledge" correlation between selected parameter distributions, propagate these uncertainties through a Monte Carlo quantification, and calculate the estimated CDF and LERF distributions.
QU-E4	Revision 1, Addendum A of the ASME/ANS Standard rewords this SR. Additionally, RG 1.200, Revision 2, provides clarification to remove Note 1.	Meets: The BVPS PRA models identify sources of model uncertainty and their related assumptions, as well as how the PRA model is affected by these.
Flooding SRs: IFPP-B1, B2, B3, IFSO-B1, B2, B3, IFSN-B1, B2, B3, IFEV-B1, B2, B3, and IFQU-B1, B2, B3.	These are new requirements for flooding that expand on the original SRs in the ASME/ANS PRA Standard.	Not Applicable (N/A): The BVPS Internal Flooding PRA models were evaluated against Part 3 of the ASME/ANS-Ra-2009 and RG 1.200, Revision 2 during the 2011 focused scope peer review. Therefore, no re-evaluation is required.
IFSN-A6	RG 1.200, Revision 2, provides clarification that should be evaluated.	N/A: The BVPS Internal Flooding PRA models were evaluated against Part 3 of the ASME/ANS-Ra-2009 and RG 1.200, Revision 2 during the 2011 focused scope peer review. Therefore, no re-evaluation is required.

3. Supporting Requirements that May Require Re-evaluation (NEI 05-04 Section 3.3.2)

   A number of the SRs changed in the ASME/ANS PRA standard as a result of the NRC comments to remove the word “key” with respect to assumptions and sources of (modeling) uncertainty.

The NEI guidance suggests that if the original peer review or self-assessment did evaluate the PRA against these NRC recommended wording changes, but the SR was assessed as "Not Met," then it may be useful for the gap assessment to include a re-evaluation of these 11 impacted SRs once the methods are modified per the disposition of the applicable F&O. The assessment of these affected SRs is provided in Table 2.

Table 2. Supporting Requirements Affected by "Key" Assumptions and Uncertainty Requiring Gap Assessment Re-evaluation

ASME/ANS RA-Sa-2009 Supporting Requirement	2007 BVPS PRA Self-Assessment / HRA Focused Peer Review SR Capability Category	Associated F&O	RG 1.200 Rev 1 to Rev 2 Gap Re-Evaluation
IE-D3	NOT MET	QU-F4-01, HR-I3-01	Meets: Note 1
AS-C3	NOT MET	HR-I3-01	Meets: Note 1
SC-C3	NOT MET	HR-I3-01	Meets: Note 1
SY-C3	Meets	None	Meets: Previously assessed as "Meets." No re-evaluation required.
HR-I3	NOT MET	HR-I3-01, HR-PR-005	Meets: Note 1
DA-E3	Meets	None	Meets: Previously assessed as "Meets." No re-evaluation required.
QU-E1	Meets	None	Meets: Previously assessed as "Meets." No re-evaluation required.
QU-E2	Meets	None	Meets: Previously assessed as "Meets." No re-evaluation required.
QU-F4	NOT MET	QU-F4-01, QU-F4-02, HR-I3-01	Meets: Note 1. Limitations in the quantification process are documented in Appendix A of the quantification notebooks.

Table 2. Supporting Requirements Affected by "Key" Assumptions and Uncertainty Requiring Gap Assessment Re-evaluation (Continued)

ASME/ANS RA-Sa-2009 Supporting Requirement	2007 BVPS PRA Self-Assessment / HRA Focused Peer Review SR Capability Category	Associated F&O	RG 1.200 Rev 1 to Rev 2 Gap Re-Evaluation
LE-D6 (LE-D5 in ASME RA-Sb-2005)	CC-1	LE-D5-01	Meets: The BVPS PRA models were updated utilizing EPRI TR-107623, "Steam Generator Tube Integrity Risk Assessment, Volume 1" as guidance. Assumptions and uncertainties were justified in this document, as well as in the performance of sensitivity studies of the thermal- and pressure-induced steam generator tube rupture values.
LE-G4	NOT MET	HR-I3-01	Meets: Note 1

Note 1: Since the BVPS internal events PRA models were peer reviewed to ASME RA-Sb-2005, as amplified by RG 1.200, Revision 1, the F&Os associated with SRs that were not met, were resolved by documenting the model uncertainty and related assumptions using the guidance of RG 1.200, Revision 1, NUREG-1855, "Guidance for the Treatment of Uncertainties Associated with PRAs in Risk Informed Decision Making," and EPRI-TR-1016737, "Treatment of Parameter and Modeling Uncertainty for Probabilistic Risk Assessments." The selection of the sources of modeling uncertainty and related assumptions were identified by those that had the most potential to change the risk metrics (for example, CDF and LERF) in a significant manner and not on a decision being made using the PRA (for example, for applications), which would be labeled as "key." As such, this SR is also compliant with RG 1.200, Revision 2, since there were no objections to these SRs.

Since the 2007 BVPS PRA self-assessment and 2007 BVPS HRA focused peer review of the internal events PRA models were reviewed against the RG 1.200, Revision 1 clarifications, the remainder of the ASME RA-Sb-2005 SRs affected by the RG 1.200, Revision 1, clarifications do not require an additional re-evaluation in a gap assessment.

4. SRs Not Requiring Re-evaluation (NEI 05-04 Section 3.3.3)

A number of the SRs changed between Addendum B (ASME RA-Sb-2005) and Revision 1, Addendum A of the ASME/ANS PRA standard (ASME/ANS RA-Sa-

2009), which do not require re-evaluation during a gap assessment. These include the numbering changes to the SRs and minor editorial changes. NEI 05-04 Rev. 3, Appendix F provides a cross-reference table of the SR numbering changes.

5. Conclusions

There were some editorial revisions and clarifications to the internal events PRA standard from the 2005 version to Part 2 "Internal Events" of the 2009 combined standard. The NRC, in RG 1.200, Revision 2, endorsed this combined standard and did not identify any exceptions. The internal events supporting requirements are essentially the same in the two standards since there are no substantive technical changes to the internal events PRA standard. This along with the NEI 05-04 Section 3.3 gap assessment provided above for the BVPS internal events PRA models, provides the basis that the BVPS internal events and internal flooding PRA models are also fully compliant with RG 1.200, Revision 2, Capability Category II or better. Therefore, the BVPS internal events PRAs based on RG 1.200, Revision 1, also conform to RG 1.200, Revision 2, and use of the current BVPS PRA models to perform the ILRT extension risk assessment would have no impact on this application.

The current BVPS seismic and internal fire PRA models have not undergone a PRA peer review against the requirements of the 2009 ASME/ANS PRA standard or RG 1.200, Revision 2, but have been subject to independent review by external events experts during the IPEEE submittal evaluation process, including the NRC and their contractors (Brookhaven National Laboratory and Sandia National Laboratory). These models are maintained in the current PRA model of record. Therefore, in accordance with the external event guidance provided in EPRI TR-1009325, Revision 2-A, Section 4.2.7, the BVPS seismic and internal fire PRA models are believed to be of sufficient scope to provide a suitable quantitative estimate of the contribution of the seismic and internal fire risk associated with this risk-informed application using a similar approach to that, which was used to calculate the change in LERF for the internal events using the guidance in EPRI TR-1 009325, Revision 2-A, Sections 4.3 and 5.1.1.

APLA RAI 2:

In the SER for the EPRI TR-1009325, Revision 2, the NRC staff stated, in part, that for licensee requests for a permanent extension of the ILRT surveillance interval to 15 years "[c]apability category I of ASME RA-Sa-2003 shall be applied as the standard, since approximate values of CDF and LERF and their distribution among release categories are sufficient for use in the EPRI methodology."

In Table 2 of Attachments 5 and 6 to the LAR, the licensee provided a "brief summary of the BVPS final resolutions to all of the 2007 BVPS PRA Self-Assessment, 2007 BVPS HRA [human reliability analysis] Focused Peer Review, and the 2011 BVPS Internal Flood PRA Focused Peer Review Facts and Observations (F&Os), which resulted in a change to the PRA model."

The licensee further stated, in part, that:

> **[a]ll other F&Os from these assessment/reviews were considered to be documentation issues, and did not impact the PRA models.**

Provide a list of all findings from the past peer-reviews and self-assessments (including the assessment in response to Question 1) relevant to this submittal for which the PRA did not meet the ASME/ANS PRA Standard capability category 1 supporting requirements. Summarize why not meeting each capability category 1 requirement will have no impact on this application.

Response:

The current BVPS-1 and BVPS-2 level 1 and level 2 PRA models that were used to perform the plant-specific risk assessments for the ILRT extensions, have been updated to meet Capability Category II or better for the supporting requirements of ASME PRA standard RA-Sb-2005 and Regulatory Guide 1.200, Revision 1 for internal events, and of the combined ASME/ANS PRA standard RA-Sa-2009 and Regulatory Guide 1.200, Revision 2 for internal flooding. Since the BVPS internal events PRAs based on RG 1.200, Revision 1, also conform to RG 1.200, Revision 2 for the internal events PRA (as discussed above in the response to APLA RAI 1), all internal events and internal flooding supporting requirements are considered to meet Capability Category II or better of the ASME/ANS PRA Standard.

An analysis of the adequacy of the internal events PRA models for both BVPS-1 and BVPS-2 (that included all the F&Os from past PRA peer reviews, focused scope peer reviews, and self-assessments and final resolution of the F&Os to meet Capability Category II or better) was previously provided to the NRC on February 14, 2014 as "Supplemental Information Regarding Application for License Amendment to Adopt National Fire Protection Association (NFPA) Standard NFPA 805, 'Performance Based Standard for Fire Protection for Light Water Reactor Electric Generating Plants (2001 Edition)'" (ADAMS Accession Number ML14051A499). The analysis includes the final resolution of F&Os that did not meet Capability Category I. Since the F&Os have been resolved to meet Capability Category II or better, these findings will have no impact on this application.

APLA RAI 3:

EPRI TR-1009325, Revision 2-A states that "[w]here possible, the analysis should include a quantitative assessment of the contribution of external events (for example, fire and seismic) in the risk impact assessment for extended ILRT intervals. For example, where a licensee possesses a quantitative fire analysis and that analysis is of sufficient quality and detail to assess the impact, the methods used to obtain the impact from internal events should be applied for the external event." For the ILRT extension request, the NRC staff can use RG 1.200, Revision 2, to determine the quality and detail of fire analyses. RG 1.200,

Revision 2, endorses a peer review using the ASME/ANS-Sa-2009 Standard with comments and clarifications.

(a) **Given that peer-reviewed Fire PRA (FPRA) models of BVPS-1 and BVPS-2 have been used in the LAR to adopt National Fire Protection Association Standard 805, "Performance-Based Standard for Fire Protection for Light Water Reactor Electric Generating Plants, 2001 Edition," on December 23, 2013 (ADAMS Accession No. ML14002A086), discuss the reasons that these peer-reviewed FPRA models were not used to estimate the contribution of fire to LERF, core damage frequency (CDF), conditional containment failure probability (CCFP) and increase in the total population dose for this application.**

Response:

FENOC understands that the EPRI guidance states that, where possible, use a quantitative assessment of the contribution of external events in the risk impact assessment for extended ILRT intervals. Although we do have RG 1.200, Revision 2 peer-reviewed FPRA models developed in support of NFPA 805 at both units, the modifications credited in those PRA models are not currently installed in the plant, so they do not represent the as-built, as-operated plants. Additionally, these FPRA models are under NRC review and are expected to change in the next year due to new research insights, NRC comments following the audit, and final modifications to resolve any unidentified issues. Simply removing the credited modifications is also not a viable option as the analysis performed was based on the proposed modifications. Removing them would again invalidate the PRA models. Therefore, we do not feel it prudent to use the NFPA 805 Fire PRA models to support this ILRT extension application.

The current BVPS-1 and BVPS-2 PRA models do have the dominant external events (seismic and internal fire) modeled using detailed full-scope level 2 PRA models, based on IPEEE methodology. These models have not undergone a PRA peer review against the requirements of the 2009 ASME/ANS PRA standard or RG 1.200, Revision 2, but have been subject to independent review by external events experts during the IPEEE submittal evaluation process, including the NRC and their contractors (Brookhaven National Laboratory and Sandia National Laboratory). These models were found to meet the requirements of Generic Letter 88-20, "Individual Plant Examination for Severe Accident Vulnerabilities – 10 CFR 50.54(f)" and NUREG-1407, "Procedural and Submittal Guidance for the Individual Plant Examination of External Events (IPEEE) for Severe Accident Vulnerabilities," to systematically and successively evaluate the seismic and fire hazards and their associated risks. Additionally, they are fully integrated with the internal events and internal flooding PRA models and are maintained in the current PRA model-of-record. As a result, the plant response modeling (fault trees and event trees) following these external initiating events has been updated as part of the PRA model update process, as well as including some fire initiating events that were originally screened-out in the IPEEE quantification. To the extent that the seismic and internal fire accident sequence logic is incorporated into the internal events PRA system event tree logic, they have also had some limited peer checks during the 2002 BVPS PRA peer review, 2007 BVPS PRA self-assessment, and 2007 BVPS HRA focused peer review. External events such as high winds, external floods, transportation, and nearby facility accidents were considered and screened-out in the IPEEE, so their risk impact is considered to be negligible compared to the impact associated with internal fires and seismic events.

Therefore, in accordance with the external event guidance provided in EPRI TR-1009325, Revision 2-A, Section 4.2.7, the BVPS seismic and internal fire PRA models are believed to be of sufficient scope to provide a suitable quantitative estimate of the contribution of the seismic and internal fire risk associated with this risk-informed application. As shown in Section 5.1 of Attachments 3 and 4 of the LAR submittal, the method chosen to account for the external events Class 3b LERF contribution is similar to the approach used to calculate the change in LERF for the internal events using the guidance in EPRI TR-1009325, Revision 2-A, Sections 4.3 and 5.1.1. This approach includes applying the NEI Interim Guidance that provided additional information concerning the conservatisms to exclude individual sequences that already independently caused LERF. However, this exclusion did not have any impact on the Class 3b frequency results, since it was more than three orders of magnitude below the total external CDF frequency.

This approach for assessing the external event contribution is similar to that used by Surry Power Station for their ILRT extension LAR, which was found to be acceptable by the NRC (ADAMS Accession Number ML14148A235).

APLA RAI 4:

In Section 3.2.2 of the SER for the EPRI TR-1009325, Revision 2, the NRC staff stated:

> **EPRI Report No. 1009325, Revision 2, also includes an assessment of the impact of the proposed change on the radiological dose to the population within a 50-mile radius of the plant. The population dose metric reflects the combined impact of the proposed change on all containment release modes/categories (including minimal, small, and large releases in both the early and late time periods), in lieu of focusing only on large early releases. This metric provides perspective on the overall impact of the proposed change on offsite consequences and ensures that these impacts will be small.**

In Section 7 of Attachment 4 to the LAR, the licensee stated:

The change in dose risk increases to 1.21 E+00 person-rem/yr when including the impact from a loss of containment overpressure (see Table 6-5). However, only 1.30E-02 person-rem/yr is attributed to the increase in the Class 3b population dose, while 1.19 person-rem/yr (2.43E+01 minus 2.31 E+01) is due to the increase in the Class 7 non-LERF frequency.

(a) Given that the reported increase in population dose rate of 1.21 person-rem/yr is larger than an acceptably small increase in population dose defined in Section 3.2.4.6 of the SER for the EPRI TR-1009325, Revision 2 (an increase in population dose of less than or equal to either 1.0 person-rem per year or 1 percent of the total population dose, whichever is less restrictive), explain why the requested ILRT extension should be accepted, even though the staff generally does not accept applications that exceed the guidelines.

(b) Provide the largest ILRT extension period at which the increase in population dose rate is less than the acceptance guideline of 1.0 person-rem/yr.

(c) As the acceptance guideline for the increase in population dose rate provides perspective on the overall impact of the proposed change on offsite consequences (including consequences from small releases in late time periods), describe the reasons that the BVPS-2 increase in population dose rate of 1.19 person-rem/yr from EPRI Accident Class 7 non-LERF (Severe accident phenomena-induced failures, non-LERF), by itself, is larger than the acceptance guideline.

Response:

(a) The sensitivity case performed for the loss of containment overpressure was a conservative analysis based on the EPRI TR-1009325, Revision 2-A guidance. This guidance suggests that as a first-order estimate of the impact, it can be assumed that the EPRI Class 3b (100 times the maximum allowable containment leakage rate specified in Technical Specifications, or 100 La) contribution would lead to loss of containment overpressure, and the systems that require this contribution for available net positive suction head (NPSH) should be made unavailable when such an isolation failure exists. As noted in Section 6.3 of Attachments 3 and 4 to the LAR, this assessment is considered to be extremely conservative since the 100 La leakage rate from the EPRI Class 3b scenarios is not likely to be of sufficient size to actually result in the loss of containment overpressure required for the recirculation spray system pumps.

Sensitivity studies have been performed for both BVPS units using the modular accident analysis program (MAAP) (reference ADAMS Accession Number ML060330262 for BVPS-1 responses to the additional questions relative to containment overpressure credit and ADAMS Accession Number ML083170522 for BVPS-2 LAR No. 08-029, Credit for Containment Overpressure) to determine the impact of operation of the most limiting recirculation spray pumps under accident conditions, with failures of containment isolation for systems that communicate directly with the containment atmosphere. The results of these studies showed that even with a 3-inch through-wall hole in containment at BVPS-1, and 2-inch through-wall hole in containment at BVPS-2, margin still existed in the NPSH available to the recirculation spray pumps.

A 2-inch equivalent diameter hole in containment also represents the size of the BVPS containment hole required for LERF, or the critical containment failure size or equivalent pipe break diameter based on a leakage rate of 100 volume percent per day. Given that the BVPS maximum allowable containment leakage rate (La) is 0.10 percent of containment air weight per day, a leakage rate of 100 percent containment air weight per day would be equivalent to 1000 La. This important aspect of LERF is also noted in NRC IMC 0308, Attachment 3, Appendix H (ADAMS Accession Number ML041340012), which states that, "Thus a LERF significant leakage rate from containment of 100% containment volume/day would correspond to about 1000 La for PWRs... The 100 volume percent per day leakage rate is approximately equivalent to a hole size in containment of 2.5-3.0 inches in diameter for PWRs with large dry containments..." Therefore, since the containment leakage rate must be greater than 1000 La in order for BVPS-1 and BVPS-2 to lose NPSH for those systems that require containment overpressure, the original analysis based on 100 La performed in Section 6.3 and Tables 6-4 and 6-5 of Attachments 3 and 4 to the LAR are considered to be overly conservative.

The EPRI TR-1009325, Revision 2-A Class 3b probability of 2.3E-03 used in the loss of containment overpressure sensitivity analysis submitted in the LAR was based on a containment leakage rate of 100 La. Therefore, the probability of having a containment leakage rate of 1000 La would be significantly less than 2.3E-03. To account for this, a scaling approach similar to that developed by the NRC to address credit for containment accident pressure in risk assessments (ADAMS Accession Number ML12234A561) was used to refine the analysis performed in Section 6.3 and revise Tables 6-4 and 6-5 of Attachments 3 and 4 to the LAR. This scaling approach was used to scale the probability of the EPRI Class 3b leakage from 100 La to 1000 La, which is now used as the minimum containment leak rate in order to result in the loss of containment overpressure available for NPSH.

The scaling approach used the expert elicitation probability (EEP) estimates taken from Table D-1 of EPRI report TR-1009325, Revision 2-A for the assumed leakage rates of 100 La and 1000 La, to scale the EPRI Class 3b base case probability of 2.3E-03 for a leak rate of 100 La based on the Jeffrey's non-informative prior (JNIP), to a leak rate of 1000 La, as follows:

$$\text{Class } 3b_{1000La} = (EEP_{1000La} / EEP_{100La}) * JNIP_{100La} = (4.50E\text{-}06 / 2.47E\text{-}04) * 2.3E\text{-}03$$

$$\text{Class } 3b_{1000La} = 0.02 * 2.3E\text{-}03 = 4.6E\text{-}05$$

Therefore, in order to lose NPSH to the pumps that take suction from the containment sump at BVPS-1 and BVPS-2 due to the loss of containment overpressure, the containment leakage rate must be at least 1000 La with an estimated probability of occurrence of 4.6E-05. Revisions to Tables 6-4 and 6-5 of

Attachments 3 and 4 to the LAR are provided to account for this refined methodology to account for the impact due to loss of containment overpressure.

As can be seen in the revised Table 6-5, the reported increase in population dose rate of 5.45E-02 person rem/ yr (or 0.14 percent) at BVPS-2 is now much smaller than an acceptably small increase in population dose defined in Section 3.2.4.6 of the SER for the EPRI TR-1009325, Revision 2 (an increase in population dose of less than or equal to either 1.0 person-rem per year or 1 percent of the total population dose, whichever is less restrictive). Therefore, the requested ILRT extension should be accepted since the application meets the guidelines.

(b) The largest ILRT extension period at which the increase in population dose rate is less than the acceptance guideline of 1.0 person-rem/yr is 15 years using the above refined methodology to estimate of the impact of the loss of containment overpressure, as shown in the revisions to Table 6.5.

(c) In lieu of discussing the BVPS-2 population dose rate increase and acceptance criteria presented in the LAR, an analysis using refined methodology (described above) has been performed to determine the increase. Using the above refined methodology to estimate of the impact of the loss of containment overpressure, the BVPS-2 increase in population dose rate from EPRI Accident Class 7 non-LERF with corrosion (Severe accident phenomena-induced failures, non-LERF) is 3.77E-02 person-rem/yr (calculated using 22.8300 person-rem/yr – 22.7923 person-rem/yr), and is much smaller than the acceptance guideline.

BVPS-1 Attachment 3 Revisions

Table 6-4. Containment Overpressure Adjustment Factors

ILRT Test Interval	Class 3b Leakage Probability Adjusted to 1000 La without Corrosion	Class 3b Leakage Probability Adjusted to 1000 La with Corrosion
3-in-10 years (Baseline)	0.02 * 2.30E-03 = 4.60E-05	0.02 * 2.30E-03 + 8.88E-06 = 5.49E-05
1-in-10 years	0.02 * 2.30E-03 * 3.33 = 1.53E-04	0.02 * 2.30E-03 * 3.33 + 5.17E-05 = 2.05E-04
1-in-15 years	0.02 * 2.30E-03 * 5.0 = 2.30E-04	0.02 * 2.30E-03 * 5.0 + 1.21E-04 = 3.51E-04

Table 6-5. BVPS-1 Loss of Containment Overpressure Total Risk for ILRT Base Case, 10, and 15 Year Extensions, Including Corrosion Impact

EPRI Class	Weighted Average Population Dose at 50 Miles (person-rem)	Base Case (3 per 10 years)					Extended to 1 per 10 years					Extended to 1 per 15 years				
		Without Corrosion		With Corrosion			Without Corrosion		With Corrosion			Without Corrosion		With Corrosion		
		Frequency (per year)	Dose Rate (person-rem per year)	Frequency (per year)	Dose Rate (person-rem per year)	Δ Dose Rate from Corrosion (person-rem per year)	Frequency (per year)	Dose Rate (person-rem per year)	Frequency (per year)	Dose Rate (person-rem per year)	Δ Dose Rate from Corrosion (person-rem per year)	Frequency (per year)	Dose Rate (person-rem per year)	Frequency (per year)	Dose Rate (person-rem per year)	Δ Dose Rate from Corrosion (person-rem per year)
1	8.00E+02	1.28E-06	1.03E-03	1.28E-06	1.03E-03	-1.64E-07	6.78E-07	5.42E-04	6.77E-07	5.41E-04	-9.94E-07	2.45E-07	1.96E-04	2.42E-07	1.94E-04	-2.55E-06
3a	8.00E+03	2.08E-07	1.66E-03	2.08E-07	1.66E-03	2.94E-08	6.92E-07	5.54E-03	6.92E-07	5.54E-03	4.92E-07	1.04E-06	8.30E-03	1.04E-06	8.31E-03	1.74E-06
3b	8.00E+04	5.19E-08	4.15E-03	5.21E-08	4.17E-03	1.61E-05	1.73E-07	1.38E-02	1.74E-07	1.39E-02	9.44E-05	2.60E-07	2.08E-02	2.62E-07	2.10E-02	2.22E-04
6	3.60E+06	9.26E-08	3.33E-01	9.26E-08	3.33E-01	7.20E-07	9.26E-08	3.33E-01	9.26E-08	3.33E-01	1.58E-05	9.26E-08	3.33E-01	9.26E-08	3.33E-01	4.64E-05
7 non-LERF	1.86E+06	1.14E-05	2.13E+01	1.14E-05	2.13E+01	7.44E-04	1.14E-05	2.13E+01	1.14E-05	2.13E+01	3.72E-03	1.14E-05	2.13E+01	1.14E-05	2.13E+01	9.13E-03
7 LERF	8.24E+06	2.49E-09	2.06E-02	2.49E-09	2.06E-02	7.42E-08	2.49E-09	2.06E-02	2.49E-09	2.06E-02	-1.76E-07	2.49E-09	2.06E-02	2.49E-09	2.06E-02	-5.18E-07
8 non-LERF	4.26E+06	9.45E-06	4.03E+01	9.45E-06	4.03E+01	0	9.45E-06	4.03E+01	9.45E-06	4.03E+01	0	9.45E-06	4.03E+01	9.45E-06	4.03E+01	0
8 LERF	4.59E+06	4.17E-08	1.91E-01	4.17E-08	1.91E-01	0	4.17E-08	1.91E-01	4.17E-08	1.91E-01	0	4.17E-08	1.91E-01	4.17E-08	1.91E-01	0
Total		2.26E-05	6.21E+01	2.26E-05	6.21E+01	7.60E-04	2.26E-05	6.22E+01	2.26E-05	6.22E+01	3.83E-03	2.26E-05	6.22E+01	2.26E-05	6.22E+01	9.39E-03
Δ Dose Rate (%Δ Dose Rate)		N/A		N/A			2.11E-02 (0.03%)		2.42E-02 (0.04%)			3.61E-02 (0.06%)		4.47E-02 (0.07%)		
CCFP		93.39%		93.39%			93.93%		93.94%			94.31%		94.33%		
Δ CCFP		N/A		N/A			0.54%		0.54%			0.92%		0.94%		
Total LERF		9.60E-08		9.62E-08			2.17E-07		2.18E-07			3.04E-07		3.06E-07		
Class 3b LERF (Δ w/Corrosion)		5.19E-08		5.21E-08 (2.01E-10)			1.73E-07		1.74E-07 (1.18E-09)			2.60E-07		2.62E-07 (2.78E-09)		
Delta LERF from Base Case [3 per 10 years] (Δ w/Corrosion)							1.21E-07		1.22E-07 (9.80E-10)			2.08E-07		2.10E-07 (2.58E-09)		
Delta CDF from Base Case [3 per 10 years] (Δ w/Corrosion)							4.11E-09		5.71E-09 (1.60E-09)			7.01E-09		1.13E-08 (4.31E-09)		

BVPS-2 Attachment 4 Revisions

Table 6-4. Containment Overpressure Adjustment Factors

ILRT Test Interval	Class 3b Leakage Probability Adjusted to 1000 La without Corrosion	Class 3b Leakage Probability Adjusted to 1000 La with Corrosion
3-in-10 years (Baseline)	0.02 * 2.30E-03 = 4.60E-05	0.02 * 2.30E-03 + 8.88E-06 = 5.49E-05
1-in-10 years	0.02 * 2.30E-03 * 3.33 = 1.53E-04	0.02 * 2.30E-03 * 3.33 + 5.17E-05 = 2.05E-04
1-in-15 years	0.02 * 2.30E-03 * 5.0 = 2.30E-04	0.02 * 2.30E-03 * 5.0 + 1.21E-04 = 3.51E-04

Table 6-5. BVPS-2 Loss of Containment Overpressure Total Risk for ILRT Base Case, 10, and 15 Year Extensions, Including Corrosion Impact

EPRI Class	Weighted Average Population Dose at 50 Miles (person-rem)	Base Case (3 per 10 years)					Extended to 1 per 10 years					Extended to 1 per 15 years				
		Without Corrosion		With Corrosion			Without Corrosion		With Corrosion			Without Corrosion		With Corrosion		
		Frequency (per year)	Dose Rate (person-rem per year)	Frequency (per year)	Dose Rate (person-rem per year)	Δ Dose Rate from Corrosion (person-rem per year)	Frequency (per year)	Dose Rate (person-rem per year)	Frequency (per year)	Dose Rate (person-rem per year)	Δ Dose Rate from Corrosion (person-rem per year)	Frequency (per year)	Dose Rate (person-rem per year)	Frequency (per year)	Dose Rate (person-rem per year)	Δ Dose Rate from Corrosion (person-rem per year)
1	8.00E+02	8.71E-07	6.97E-04	8.71E-07	6.97E-04	-2.01E-07	4.26E-07	3.41E-04	4.25E-07	3.40E-04	-7.86E-07	1.08E-07	8.68E-05	1.06E-07	8.47E-05	-2.11E-06
3a	8.00E+03	1.52E-07	1.22E-03	1.52E-07	1.22E-03	2.88E-08	5.08E-07	4.07E-03	5.08E-07	4.07E-03	8.07E-07	7.63E-07	6.10E-03	7.63E-07	6.10E-03	2.83E-06
3b	8.00E+04	3.81E-08	3.05E-03	3.83E-08	3.06E-03	1.18E-05	1.27E-07	1.02E-02	1.28E-07	1.02E-02	7.05E-05	1.91E-07	1.53E-02	1.93E-07	1.54E-02	1.67E-04
6	3.59E+06	4.00E-08	1.44E-01	4.00E-08	1.44E-01	1.08E-06	4.00E-08	1.44E-01	4.00E-08	1.44E-01	1.76E-05	4.00E-08	1.44E-01	4.00E-08	1.44E-01	3.43E-05
7 non-LERF	1.95E+06	1.17E-05	2.28E+01	1.17E-05	2.28E+01	9.54E-04	1.17E-05	2.28E+01	1.17E-05	2.28E+01	6.38E-03	1.17E-05	2.28E+01	1.17E-05	2.28E+01	1.53E-02
7 LERF	8.24E+06	1.92E-09	1.58E-02	1.92E-09	1.58E-02	0	1.92E-09	1.58E-02	1.92E-09	1.58E-02	1.56E-07	1.92E-09	1.58E-02	1.92E-09	1.58E-02	-1.16E-08
8 non-LERF	4.26E+06	3.73E-06	1.59E+01	3.73E-06	1.59E+01	0	3.73E-06	1.59E+01	3.73E-06	1.59E+01	0	3.73E-06	1.59E+01	3.73E-06	1.59E+01	0
8 LERF	4.28E+06	1.77E-07	7.57E-01	1.77E-07	7.57E-01	0	1.77E-07	7.57E-01	1.77E-07	7.57E-01	0	1.77E-07	7.57E-01	1.77E-07	7.57E-01	0
Total		1.67E-05	3.96E+01	1.67E-05	3.96E+01	9.67E-04	1.67E-05	3.96E+01	1.67E-05	3.96E+01	6.47E-03	1.67E-05	3.97E+01	1.67E-05	3.97E+01	1.55E-02
Δ Dose Rate (%Δ Dose Rate)		N/A		N/A			2.34E-02 (0.06%)		2.89E-02 (0.07%)			3.99E-02 (0.10%)		5.45E-02 (0.14%)		
CCFP		93.88%		93.88%			94.41%		94.42%			94.79%		94.81%		
Δ CCFP		N/A		N/A			0.54%		0.54%			0.92%		0.93%		
Total LERF		2.17E-07		2.17E-07			3.06E-07		3.07E-07			3.70E-07		3.72E-07		
Class 3b LERF (Δ w/Corrosion)		3.81E-08		3.83E-08 (1.48E-10)			1.27E-07		1.28E-07 (8.81E-10)			1.91E-07		1.93E-07 (2.09E-09)		
Delta LERF from Base Case [3 per 10 years] (D w/Corrosion)							8.90E-08		8.97E-08 (7.33E-10)			1.53E-07		1.55E-07 (1.94E-09)		
Delta CDF from Base Case [3 per 10 years] (D w/Corrosion)							6.87E-09		9.76E-09 (2.89E-09)			1.18E-08		1.91E-08 (7.28E-09)		

APLA RAI 5:

In Section 5.1 of Attachments 3 and 4 to the LAR, the licensee states, in part, that:

> **a containment isolation analysis was performed to estimate the frequency of failure to isolate lines that could cause a significant risk of radioactive release. The results of this analysis screened-out all containment penetrations > 2-inch diameter. Furthermore, the Beaver Valley Power Station containments are operated at slightly sub-atmospheric pressures (BVPS Technical Specification LCO 3.6.4 states that containment pressure shall be $\geq$ 12.8 psia and $\leq$ 14.2 psia), thus the baseline PRA models do not consider a large pre-existing loss of containment isolation to be credible. Therefore, the frequency per year for these Class 2 sequences is assumed to be zero.**

Section 4.3 of EPRI TR-1009325, Revision 2, provides guidance for calculating frequency per year for these sequences as follows:

$$F_{\text{Class 2}} = PROB_{\text{large CI}} * CDF_{\text{Total}}$$

Where:

$PROB_{\text{large CI}}$ = random containment large isolation failure probability (large valves), and

CDF_{Total} = total plant-specific core damage frequency, which is obtained from plant specific PRA.

(a) By the letter dated February 6, 2006 (ADAMS Accession No. ML060100325), NRC approved conversion of BVPS-1 and BVPS-2 containments from sub-atmospheric to atmospheric operating condition. Discuss the impacts of operating under atmospheric condition on this application.

(b) As the LAR, submitted by letter dated August 12, 2013 [ADAMS Accession No. ML13232A042], for extension of the Type A test interval for similar sub-atmospheric containments at Surry Power Station did not assume a zero frequency per year for these Class 2 sequences, describe the screening criteria used to analyze containment penetrations > 2-inch diameter that resulted in screening out all these containment penetrations.

(c) Using the guidance in EPRI TR-1009325, Revision 2, calculate $F_{\text{Class 2}}$ and show how the calculated $F_{\text{Class 2}}$ will change the plant specific risk assessment results (reported in Table 6-5 of Attachments 3 and 4 to the LAR). If $PROB_{\text{large CI}}$ is assumed to be zero, explain why this failure probability is zero.

Response:

(a) Although the NRC approved conversion of the BVPS-1 and BVPS-2 containments from sub-atmospheric to atmospheric operating conditions, the BVPS containments are operated at slightly sub-atmospheric pressures (typically about 13.5 pounds per square inch absolute [psia], where 14.3 psia is the referenced atmospheric pressure for design basis accident analyses). Operation at atmospheric conditions is prohibited by BVPS Technical Specification, Limiting Condition for Operation 3.6.4, which states that containment pressure shall be greater than or equal to 12.8 psia and less than or equal to 14.2 psia. Operation at atmospheric conditions would place BVPS outside of its design basis, since the containment pressure is limited during normal operation to preserve the initial conditions assumed in the accident analyses for a loss of coolant accident (LOCA) or steam line break (SLB). Maintaining containment pressure at less than or equal to the limiting condition for operation upper pressure limit ensures that, in the event of a design basis accident, the resultant peak containment pressure will remain below the containment design pressure. Section 7 of Attachments 3 and 4 to the LAR states that Technical Specification Surveillance Requirement 3.6.4.1 verifies that containment pressure is within the above-mentioned limits every 12 hours. This 12-hour frequency is considered adequate in view of other indications available in the control room, including alarms, to alert the operator to an abnormal containment pressure condition.

Therefore, there are no impacts of operating at atmospheric conditions on this application, since doing so would place the BVPS units into a 1-hour completion time to restore containment pressure to within limits, below atmospheric pressure conditions.

(b) The Surry Power Station ILRT extension analysis conservatively assumed that a greater than 1-inch diameter hole in containment would lead to a large containment failure, and hence classified these as EPRI Class 2 release sequences. A greater than nominal 2-inch diameter hole in containment is used as the criterion for a large containment failure at BVPS, which is based on WCAP-16378 "Westinghouse Owners Group Definition for Large Early Release Frequency (LERF)" and NUREG-1493 "Performance-Based Containment Leak-Test Program," as the minimum containment penetration size that can result in a 100 percent containment volume per day leak rate at design pressure.

The objective of the BVPS containment isolation analysis was to estimate the frequency of failure to isolate lines that could cause a significant risk of radioactive release. Since there are too many lines to analyze individually, penetrations are screened to eliminate those that would not be risk-significant. The screening criteria have evolved to eliminate those penetrations found in previous analyses to be relatively unimportant. The screening criteria and the systems screened-out as a result of their application are as follows.

Penetrations were screened out if their lines have neither inlet nor outlet connected to either the RCS or containment environment. This criterion eliminated the following systems:

- Steam generator blowdown
- Blowdown sampling
- Component cooling
- Service/river water
- Main steam
- Main feedwater
- Auxiliary feedwater
- Steam drain
- Steam vents
- Containment instrument air supply

Penetrations were screened out if their lines were connected to the RCS or containment environment, but their containment isolation valves were required to be open for safety injection, containment depressurization, or other post-initiating event safety function. This criterion eliminated the following systems:

- Quench spray
- Recirculation spray
- Safety injection
- Seal water injection
- Leakage monitoring

Penetrations were screened out if their lines were isolated during normal power operation by a normally closed, fail-closed isolation valve; a normally closed manual valve; or at least three check valves in series. This criterion eliminated the following systems:

- Chemical and volume control (charging)
- Post-DBA [Design Basis Accident] hydrogen control
- Fire protection water
- Containment purge
- Residual heat removal
- Accumulator test line
- Post-accident sampling
- Normal sampling
- Personnel air lock
- Emergency air lock
- Fuel transfer tube
- Fuel pool cooling and purification (BVPS-2)

The systems remaining after the screening analysis:

- Vent and drain
- Fuel pool cooling and purification (BVPS-1)
- Reactor coolant
- Containment vacuum
- Gas supply

All of these system penetrations were further screened-out as being contributors to large containment isolation failures since their equivalent line sizes were not greater than a nominal 2-inch diameter. As a result, they were considered to be small containment isolation failures and were subsequently binned in the EPRI Class 6 release sequences for the BVPS ILRT extension risk assessments.

(c) As stated in the above responses, BVPS does not consider any containment penetration isolation failures to be large, therefore the $PROB_{large\ CI}$ is assumed to be zero.

SCVB RAI 1:

In accordance with Option 8- Performance-Based Requirements, Subsection *V.B. Implementation,* of 10 CFR Part 50, Appendix J, it states, in part, that, "regulatory guides or other implementation documents used by a licensee to develop a performance-based leakage testing program must be included, by general reference, in the plant technical specifications. The submittal for technical specification revisions must contain justification, including supporting analyses, if the licensee chooses to deviate from methods approved by the Commission and endorsed in a regulatory guide."

Enclosure Attachments 1 and 2 (Mark-up and Re-typed Change of Technical Specifications) of LAR include reference to NEI 94-01, Revision 3-A. However, there are certain conditions and limitations contained in NEI 94-01, Revision 2-A that were not incorporated in NEI 94-01, Revision 3-A. To accommodate for this omission, the NRC staff recommends that licensees reference in their technical specifications both NEI 94-01, Revision 3-A, and the conditions and limitations specified in NEI 94-01, Revision 2-A.

Please include a reference to the conditions and limitations of NEI 94-01, Revision 2-A, or provide justification for omitting any reference to NEI 94-01, Revision 2-A, as part of the proposed changes to Technical Specifications 5.5.12-a.

Response:

The Technical Specification 5.5.12.a wording proposed in the license amendment request submitted by letter dated April 16, 2014 is hereby revised to read as follows:

This program shall be in accordance with the guidelines contained in NEI 94-01, Revision 3-A, "Industry Guideline for Implementing Performance-Based Option of 10 CFR Part 50, Appendix J," and conditions and limitations specified in NEI 94-01, Revision 2-A.

Attachment 2 provides a copy of Technical Specification page 5.5-19 marked to show the revised wording. Text to be deleted is marked with a line through the letters, and text to be added is shown underlined. Technical Specification page 5.5-19 re-typed to present what the page will look like after the proposed changes are incorporated is provided in Attachment 3.

EMCB RAI 1:

The first five rows in the "Scheduled Outage" column of the table in Section 3.2.1, "Containment Inservice Inspection Program", on page 21 of 54 of the licensee's submittal list "1R25, 1R28, 1R30," and the 10th and 11th rows list "2R20, 2R23, 2R24."

(a) Explain why 1R27, 1R29 and 1R31 are not listed for the first five rows and 2R22 is not listed for rows 10 and 11.

(b) Also, in the same table, 4th column, "Exam Method," explain why VT-3 examination is performed once per 10-year interval for 1-CNMT-SPARE-PENE-BOLTING, and 2-CNMT-ELEC-PENE-BOLTING.

Response:

(a) ASME Boiler and Pressure Vessel Code Section XI, Subsection IWE, Table IWE-2500-1, Examination Category E-A, Item Number E1.11, indicates that 100 percent of the accessible surface areas on the containment liner are required to be examined once each 40-month inspection period. The Beaver Valley containment ISI program has scheduled this examination in the first refueling outage of each 40-month inspection period. The specific outages in question were not the first outages of the inspection period and did not require examination.

Notes:

1) There are three refueling outages in the first 40-month period for both the fifth 10-year interval at BVPS-1 and the fourth 10-year interval at BVPS-2.

2) Following submittal of this amendment request the general visual examination frequency of the BVPS-1 liner was increased to every outage in response to a condition report that identified liner corrosion in 1R22 (Fall 2013). Section 3.2.6 of the LAR contains additional details.

(b) Per 10 CFR 50.55a(b)(2)(ix)(G), an examination of the pressure-retaining bolted connections in Item E1.11 of Table IWE–2500–1 using the VT–3 examination method must be conducted once each interval. The components 1-CNMT-SPARE-PENE-BOLTING and 2-CNMT-ELEC-PENE-BOLTING referred to above in request EMCB RAI 1(b), contain pressure-retaining bolted connections.

While the VT-3 examination on pressure-retaining bolted connections will be performed once each 10-year interval, the general visual examination of the accessible interior and exterior containment surfaces will be conducted at the frequency required by NEI 94-01, Revision 3A. The general visual examination includes surface areas of the liner plate, welds, integral attachments, containment penetrations, pressure retaining bolted connections and the moisture barrier. The frequency of general visual examination of the BVPS-1 containment liner was increased as a corrective action from the identified liner corrosion in 1R22 (Fall 2013). Exterior surface examination of the BVPS-1 containment will be scheduled at the frequency listed in NEI 94-01, Revision 3A. General visual examination of the BVPS-2 containment will be scheduled prior to each Type A test and at least three other outages before the next Type A test if the interval has been extended to 15 years.

EMCB RAI 2:

Section 3.1.2, "Type B and C Testing," and Tables on pages 12 and 13 of 54 of Reference 1 to the LAR, describes Type B and C testing and leak rate summation history for Units 1 and 2 (respectively), which indicated the as-found minimum pathway summations, and the as-left maximum pathway summations, representing the effective management of the Containment Leakage Rate Testing Program. In order for the NRC staff to assess the proper and effective implementation of the Type B and Type C local leak rate testing program for each unit, please provide the following:

(a) For the last two consecutive periodic tests, please provide a table that has: (1) the component(s) that have not demonstrated acceptable performance; (2) the as-found value; (3) the acceptable value; (4) the as-left value; (5) the causes of the test failure; (6) the corrective actions taken; and (7) the next test schedule intervals.

(b) A discussion of any operating experience and evaluation results, regarding the potential for, or presence of, corrosive conditions at the concrete-to-metal interface at the basement floor of the containment. The discussion should include the potential for stagnant water to be trapped behind a degraded floor seal (moisture barrier) area that could promote pitting corrosion. This item is requested to be consistent with NRC Information Notice 2004-09, "Corrosion of Steel Containment and Containment Liner."

Response:

(a) The following tables identify the components that have not demonstrated acceptable performance during the last two consecutive periodic tests at each unit. As-found and as-left values have been rounded to the nearest standard cubic foot per day (SCF/D).

Table 3. 1R21 Components

Component	As-found SCF/D	Admin Limit SCF/D	As-left SCF/D	Cause of Failure	Corrective Action	Scheduled Interval
Electrical Penetration 1RCP-3F Canister	54	3.0	62	Source of leakage could not be located – suspected at wire-penetration interface	Replacement penetration was ordered and work order generated for replacement during 1R22	30 months
TV-1RC-101	135	80	4	Debris on seat	Valve was disassembled, cleaned, inspected and repacked	30 months

Table 4. 1R22 Components

Component	As-found SCF/D	Admin Limit SCF/D	As-left SCF/D	Cause of Failure	Corrective Action	Scheduled Interval
Electrical Penetration 1RCP-3F Canister	47	3.0	0.05	Source of leakage could not be located or repaired – suspected at wire-penetration interface	Replacement penetration installed and tested SAT	30 months

Table 5. 2R16 Components

Component	As-found SCF/D	Admin Limit SCF/D	As-left SCF/D	Cause of Failure	Corrective Action	Scheduled Interval
2CVS-93	219	80	7	1/2 inch. area of adhesive was found on the poppet surface.	Poppet and seating surfaces were cleaned	Permanent 30 month schedule previously established

Table 6. 2R17 Components

Component	As-found SCF/D	Admin Limit SCF/D	As-left SCF/D	Cause of Failure	Corrective Action	Scheduled Interval
2SAS-14	208	150	208	Leakage evaluated and accepted into containment leakage rate Repair planned for next refueling outage	Corrective Action has not been determined – will be determined following investigation in the next refueling outage Investigation / Repair order added to outage scope	30 months
2CVS-93	2432	80	3	Significant amount of debris (dirt) found on the seating surface	Seating surface was cleaned and 360 degree seating check was performed SAT	Permanent 30 month schedule previously established

(b) Operating experience and evaluation results, regarding the potential for, or presence of, corrosive conditions at the concrete-to-metal interface at the basement floor of the containment is described below.

During the BVPS-1 2000 refueling outage, rust on the liner was identified where the wall meets the floor in the vicinity of the containment sump (left and right of the sump where the floor slopes to the sump). The moisture barrier for the reactor containment had a 17-inch separation in the caulking to the concrete along the floor to wall interface (less than 1/32 inch wide). There was surface corrosion but negligible metal thinning. The caulking was replaced.

During the BVPS-1 2003 refueling outage, deteriorated caulking was identified at two areas of the wall/floor interface. Chipped paint was identified on the liner in the vicinity of the containment sump (elevation 692 foot). The deteriorated caulking was removed and recaulked. No liner corrosion was identified.

During the BVPS-1 2010 refueling outage, degraded caulking at the wall to floor intersection was identified. No liner corrosion was identified, and the caulking was repaired.

During the BVPS-1 2012 refueling outage, degraded caulking at the wall to floor intersection was identified. No liner corrosion was identified, and the caulking was replaced.

During the BVPS-1 2013 refueling outage, a containment coatings inspection was performed on the interior containment liner, and an indication at elevation 692 foot approximately 7 inches from the floor was identified. Section 3.2.6 of the LAR details the evaluation.

During the BVPS-2 2006 refueling outage, grouting was removed from the wall/floor interface for the containment sump modification (elevation 692 foot). Previously unpainted liner plate was exposed during concrete removal for the containment sump modification (elevation 692 foot). No liner corrosion was identified. The wall/floor interface was restored.

During the BVPS-2 2008 refueling outage, numerous rust spots at floor/wall interface around the containment (elevation 692 foot) were identified. The areas were cleaned and repainted. There was surface corrosion but negligible metal thinning.

Examination and maintenance of the moisture barrier at the containment floor to wall interface has been performed since construction. While surface corrosion has been observed, there has been negligible liner degradation associated with the moisture barrier as discussed in Information Notice 2004-09.

Attachment 2
L-14-343

Proposed Technical Specification Change (Mark-up)
(1 page follows)

5.5 Programs and Manuals

5.5.11 Safety Function Determination Program (SFDP) (continued)

c. The SFDP identifies where a loss of safety function exists. If a loss of safety function is determined to exist by this program, the appropriate Conditions and Required Actions of the LCO in which the loss of safety function exists are required to be entered. When a loss of safety function is caused by the inoperability of a single Technical Specification support system, the appropriate Conditions and Required Actions to enter are those of the support system.

5.5.12 Containment Leakage Rate Testing Program

a. A program shall establish the leakage rate testing of the containment as required by 10 CFR 50.54(o) and 10 CFR 50, Appendix J, Option B, as modified by approved exemptions. For Unit 1, exemptions to Appendix J of 10 CFR 50 are dated November 19, 1984, ~~December 5, 1984,~~ and July 26, 1995. For Unit 2, exemptions to Appendix J of 10 CFR 50 are as stated in the Operating License. This program shall be in accordance with the guidelines contained in ~~Regulatory Guide 1.163, "Performance-Based Containment Leak-Test Program," dated September, 1995, as modified by the following exceptions:~~

~~1. For Unit 1, the next Type A test performed after the May 29, 1993 Type A test shall be performed no later than May 28, 2008.~~

~~2. For Unit 2, the next Type A test performed after the November 10, 1993 Type A test shall be performed no later than November 9, 2008.~~

NEI 94-01, Revision 3-A, "Industry Guideline for Implementing Performance-Based Option of 10 CFR 50, Appendix J," and conditions and limitations specified in NEI 94-01, Revision 2-A.

b. The calculated peak containment internal pressure for the design basis loss of coolant accident, P_a, is 43.1 psig (for Unit 1) and 44.8 psig (for Unit 2).

c. The maximum allowable containment leakage rate, L_a, at P_a, shall be 0.10% of containment air weight per day.

d. Leakage rate acceptance criteria are:

1. Containment leakage rate acceptance criterion is $\leq$ 1.0 L_a. However, during the first unit startup prior to MODE 4 entry following testing in accordance with this program, the leakage rate acceptance criteria are < 0.60 L_a for the Type B and C tests and $\leq$ 0.75 L_a for Type A tests.

2. Air lock testing acceptance criteria are:

   a) Overall air lock leakage rate is $\leq$ 0.05 L_a when tested at $\geq$ P_a.

Attachment 3
L-14-343

Proposed Technical Specification Change (Re-Typed)
(1 page follows)

5.5 Programs and Manuals

5.5.11 Safety Function Determination Program (SFDP) (continued)

c. The SFDP identifies where a loss of safety function exists. If a loss of safety function is determined to exist by this program, the appropriate Conditions and Required Actions of the LCO in which the loss of safety function exists are required to be entered. When a loss of safety function is caused by the inoperability of a single Technical Specification support system, the appropriate Conditions and Required Actions to enter are those of the support system.

5.5.12 Containment Leakage Rate Testing Program

a. A program shall establish the leakage rate testing of the containment as required by 10 CFR 50.54(o) and 10 CFR 50, Appendix J, Option B, as modified by approved exemptions. For Unit 1, exemptions to Appendix J of 10 CFR 50 are dated November 19, 1984, and July 26, 1995. For Unit 2, exemptions to Appendix J of 10 CFR 50 are as stated in the Operating License. This program shall be in accordance with the guidelines contained in NEI 94-01, Revision 3-A, "Industry Guideline for Implementing Performance-Based Option of 10 CFR 50, Appendix J," and conditions and limitations specified in NEI 94-01, Revision 2-A.

b. The calculated peak containment internal pressure for the design basis loss of coolant accident, P_a, is 43.1 psig (for Unit 1) and 44.8 psig (for Unit 2).

c. The maximum allowable containment leakage rate, L_a, at P_a, shall be 0.10% of containment air weight per day.

d. Leakage rate acceptance criteria are:

1. Containment leakage rate acceptance criterion is $\leq 1.0\ L_a$. However, during the first unit startup prior to MODE 4 entry following testing in accordance with this program, the leakage rate acceptance criteria are $< 0.60\ L_a$ for the Type B and C tests and $\leq 0.75\ L_a$ for Type A tests.

2. Air lock testing acceptance criteria are:

   a) Overall air lock leakage rate is $\leq 0.05\ L_a$ when tested at $\geq P_a$.