

ATTACHMENT (3)

**EVALUATION OF RISK SIGNIFICANCE OF PERMANENT ILRT
EXTENSION**

Calvert Cliffs Nuclear Power Plant: Evaluation of Risk Significance of Permanent ILRT Extension

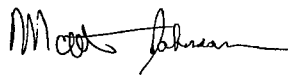
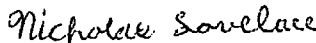

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REVISION RECORD SUMMARY

Revision	Revision Summary
0	Initial Issue
1	Incorporated True North review comments.
2	Incorporated minor comments regarding NFPA 805 transition in Section 5.1.2.
3	Removed generic QA condition statement and generic containment overpressure discussion in section 2.0.
4	<p>Incorporated RAI responses. Edited Table 1 of the attachment to indicate and discuss which Supporting Requirements were "not met." Added Tables 5-33 - 5-36 with CCFP and annual dose rates for the steel liner corrosion sensitivities. Added phrases in Section 5.3.2 to introduce these tables. Updated Tables 5-25, 5-26, and 5-27 and other calculated values in Section 5.3.1 to reflect the Seismic CDF values given in Section 8 of the IPEEE report for both units. Updated Table 5-28, 5-29, and 5-30 and other calculated values in Section 5.3.1.1 to reflect the Unit 2 Fire CDF value given in Section 8 of the IPEEE. Added additional discussion to Section 5.2.4 to clarify the release timing. Added Attachment 2, which contains the release timing plots from NC-94-020 [Reference 18] for MAAP cases HRIF, GIOY, and MRIF. Revised Section 5.1.4 to include disposition of recent corrosion data. Revised section 5.3.2 to clarify that the dose increase estimate is based on the class 3b contribution with containment spray success and late sequences removed. Added Attachment 3 as placeholder for RAI responses.</p>

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1.0 PURPOSE

The purpose of this analysis is to provide a risk assessment of extending the currently allowed containment Type A Integrated Leak Rate Test (ILRT) to permanent fifteen years. The extension would allow for substantial cost savings as the ILRT could be deferred for additional scheduled refueling outages for the Calvert Cliffs Nuclear Power Plant (CCNPP). The risk assessment follows the guidelines from NEI 94-01, Revision 3-A [Reference 1], the methodology used in EPRI TR-104285 [Reference 2], the NEI "Interim Guidance for Performing Risk Impact Assessments in Support of One-Time Extensions for Containment Integrated Leakage Rate Test Surveillance Intervals" from November 2001 [Reference 3], the NRC regulatory guidance on the use of Probabilistic Risk Assessment (PRA) as stated in Regulatory Guide 1.200 as applied to ILRT interval extensions, and risk insights in support of a request for a plant's licensing basis as outlined in Regulatory Guide (RG) 1.174 [Reference 4], the methodology used for Calvert Cliffs to estimate the likelihood and risk implications of corrosion-induced leakage of steel liners going undetected during the extended test interval [Reference 5], the methodology used in EPRI 1009325, Revision 2-A [Reference 24], and the methodology improvements in EPRI 1018243 [Reference 24].

2.0 SCOPE

Revisions to 10CFR50, Appendix J (Option B) allow individual plants to extend the Integrated Leak Rate Test (ILRT) Type A surveillance testing frequency requirement from three in ten years to at least once in ten years. The revised Type A frequency is based on an acceptable performance history defined as two consecutive periodic Type A tests at least 24 months apart in which the calculated performance leakage rate was less than limiting containment leakage rate of 1La.

The basis for the current 10-year test interval is provided in Section 11.0 of NEI 94-01, Revision 0, and established in 1995 during development of the performance-based Option B to Appendix J. Section 11.0 of NEI 94-01 states that NUREG-1493, "Performance-Based Containment Leak Test Program," September 1995 [Reference 6], provides the technical basis to support rulemaking to revise leakage rate testing requirements contained in Option B to Appendix J. The basis consisted of qualitative and quantitative assessment of the risk impact (in terms of increased public dose) associated with a range of extended leakage rate test intervals. To supplement the NRC's rulemaking basis, NEI undertook a similar study. The results of that study are documented in Electric Power Research Institute (EPRI) Research Project TR-104285, "Risk Impact Assessment of Revised Containment Leak Rate Testing Intervals".

The NRC report on performance-based leak testing, NUREG-1493, analyzed the effects of containment leakage on the health and safety of the public and the benefits realized from the containment leak rate testing. In that analysis, it was determined that for a representative PWR plant (i.e., Surry), that containment isolation failures contribute less than 0.1 percent to the latent risks from reactor accidents. Consequently, it is desirable to show that extending the ILRT interval will not lead to a substantial increase in risk from containment isolation failures for CCNPP.

NEI 94-01 Revision 2-A contains a Safety Evaluation Report that supports using EPRI Report No. 1009325 Revision 2-A, *Risk Impact Assessment of Extended Integrated Leak Rate Testing Intervals*, for performing risk impact assessments in support of ILRT extensions [Reference 24]. The Guidance provided in Appendix H of EPRI Report No. 1009325 Revision 2-A builds on the EPRI Risk Assessment methodology, EPRI TR-104285. This methodology is followed to determine the appropriate risk information for use in evaluating the impact of the proposed ILRT changes.

It should be noted that containment leak-tight integrity is also verified through periodic in-service inspections conducted in accordance with the requirements of the American Society of Mechanical Engineers (ASME) Boiler and Pressure Vessel Code Section XI. More specifically, Subsection IWE provides the rules and requirements for in-service inspection of Class MC pressure-retaining components and their integral attachments, and of metallic shell and penetration liners of Class CC pressure-retaining components and their integral attachments in light-water cooled plants. Furthermore, NRC regulations 10 CFR 50.55a(b)(2)(ix)(E) require licensees to conduct visual inspections of the accessible areas of the interior of the containment. The associated change to NEI 94-01 will require that visual examinations be conducted during at least three other outages, and in the outage during which the ILRT is being conducted. These requirements will not be changed as a result of the extended ILRT interval. In addition, Appendix J, Type B local leak tests performed to verify the leak-tight integrity of containment penetration bellows, airlocks, seals, and gaskets are also not affected by the change to the Type A test frequency.

The acceptance guidelines in RG 1.174 are used to assess the acceptability of this permanent extension of the Type A test interval beyond that established during the Option B rulemaking of Appendix J. RG 1.174 defines very small changes in the risk-acceptance guidelines as increases in Core Damage Frequency (CDF) less than 10^{-6} per reactor year and increases in Large Early Release Frequency (LERF) less than 10^{-7} per reactor year. Since the Type A test does not impact CDF, the relevant criterion is the change in LERF. RG 1.174 also defines small changes in LERF as below 10^{-6} per reactor year. RG 1.174 discusses defense-in-depth and encourages the use of risk analysis techniques to help ensure and show that key principles, such as the defense-in-depth philosophy, are met. Therefore, the increase in the Conditional Containment Failure Probability (CCFP), which helps ensure the defense-in-depth philosophy is maintained, is also calculated.

Regarding CCFP, changes of up to 1.1% have been accepted by the NRC for the one-time requests for extension of ILRT intervals. In context, it is noted that a CCFP of 1/10 (10%) has been approved for application to evolutionary light water designs. Given these perspectives, a change in the CCFP of up to 1.5% is assumed to be small.

In addition, the total annual risk (person rem/year population dose) is examined to demonstrate the relative change in this parameter. While no acceptance guidelines for these additional figures of merit are published, examinations of NUREG-1493 and Safety Evaluation Reports (SER) for one-time interval extension (summarized in Appendix G) indicate a range of incremental increases in population dose that have been accepted by the NRC. The range of incremental population dose increases is from ≤ 0.01 to 0.2 person-rem/year and/or 0.002% to 0.46% of the total accident dose. The total doses for the spectrum of all accidents (NUREG-1493 [Reference 6], Figure 7-2) result in health effects that are at least two orders of magnitude less than the NRC Safety Goal Risk. Given these perspectives, a very small population dose is defined as an increase from the baseline interval (3 tests per 10 years) dose of ≤ 1.0 person-rem per year or 1% of the total baseline dose, whichever is less restrictive for the risk impact assessment of the proposed extended ILRT interval.

For those plants that credit containment overpressure for the mitigation of design basis accidents, a brief description of whether overpressure is required should be included in this section. In addition, if overpressure is included in the assessment, other risk metrics such as CDF should be described and reported.

3.0 REFERENCES

The following references were used in this calculation:

1. *Revision 3-A to Industry Guideline for Implementing Performance-Based Option of 10 CFR Part 50, Appendix J*, NEI 94-01, July 2012.
2. *Risk Impact Assessment of Revised Containment Leak Rate Testing Intervals*, EPRI, Palo Alto, CA EPRI TR-104285, August 1994.
3. *Interim Guidance for Performing Risk Impact Assessments in Support of One-Time Extensions for Containment Integrated Leakage Rate Test Surveillance Intervals*, Revision 4, developed for NEI by EPRI and Data Systems and Solutions, November 2001.
4. *An Approach for Using Probabilistic Risk Assessment in Risk-Informed Decisions on Plant-Specific Changes to the Licensing Basis*, Regulatory Guide 1.174, May 2011.
5. *Response to Request for Additional Information Concerning the License Amendment Request for a One-Time Integrated Leakage Rate Test Extension*, Letter from Mr. C. H. Cruse (Calvert Cliffs Nuclear Power Plant) to NRC Document Control Desk, Docket No. 50-317, March 27, 2002.
6. *Performance-Based Containment Leak-Test Program*, NUREG-1493, September 1995.
7. *Evaluation of Severe Accident Risks: Surry Unit 1*, Main Report NUREG/CR-4551, SAND86-1309, Volume 3, Revision 1, Part 1, October 1990.
8. Letter from R. J. Barrett (Entergy) to U. S. Nuclear Regulatory Commission, IPN-01-007, January 18, 2001.
9. United States Nuclear Regulatory Commission, Indian Point Nuclear Generating Unit No. 3 – Issuance of Amendment Re: Frequency of Performance-Based Leakage Rate Testing (TAC No. MB0178), April 17, 2001.
10. *Impact of Containment Building Leakage on LWR Accident Risk*, Oak Ridge National Laboratory, NUREG/CR-3539, ORNL/TM-8964, April 1984.
11. *Reliability Analysis of Containment Isolation Systems*, Pacific Northwest Laboratory, NUREG/CR-4220, PNL-5432, June 1985.
12. *Technical Findings and Regulatory Analysis for Generic Safety Issue II.E.4.3 'Containment Integrity Check'*, NUREG-1273, April 1988.
13. *Review of Light Water Reactor Regulatory Requirements*, Pacific Northwest Laboratory, NUREG/CR-4330, PNL-5809, Volume 2, June 1986.
14. *Shutdown Risk Impact Assessment for Extended Containment Leakage Testing Intervals Utilizing ORAM™*, EPRI, Palo Alto, CA, TR-105189, Final Report, May 1995.
15. *Severe Accident Risks: An Assessment for Five U. S. Nuclear Power Plants*, NUREG-1150, December 1990.
16. United States Nuclear Regulatory Commission, *Reactor Safety Study*, WASH-1400, October 1975.
17. Calculation No. C0-QU-001, Revision 1, Calvert Cliffs Nuclear Power Plant, Unit 1, "PRA Quantification (QU) Notebook," July 2012.
18. Calculation No. NC-94-020, "Severe Accident Analysis of Calvert Cliffs for IPE Level II," December 1994.

19. Massoud, M., Calculation No. CA07463, Revision 0, "2010 Update of Dose Analysis for Level 3 PRA Release Categories," August 2001.
20. Anthony R. Pietrangelo, One-time extensions of containment integrated leak rate test interval – additional information, NEI letter to Administrative Points of Contact, November 30, 2001.
21. Letter from J. A. Hutton (Exelon, Peach Bottom) to U. S. Nuclear Regulatory Commission, Docket No. 50-278, License No. DPR-56, LAR-01-00430, dated May 30, 2001.
22. *Risk Assessment for Joseph M. Farley Nuclear Plant Regarding ILRT (Type A) Extension Request*, prepared for Southern Nuclear Operating Co. by ERIN Engineering and Research, P0293010002-1929-030602, March 2002.
23. Letter from D. E. Young (Florida Power, Crystal River) to U. S. Nuclear Regulatory Commission, 3F0401-11, dated April 25, 2001.
24. *Risk Impact Assessment of Extended Integrated Leak Rate Testing Intervals*, Revision 2-A of 1009325, EPRI, Palo Alto, CA. 1018243, October 2008.
25. *Risk Assessment for Vogtle Electric Generating Plant Regarding the ILRT (Type A) Extension Request*, prepared for Southern Nuclear Operating Co. by ERIN Engineering and Research, February 2003.
26. *Perspectives Gained from the IPEEE Program*, USNRC, NUREG-1742, April 2002.
27. Procedure STP M-662-1, Revision 6, Calvert Cliffs Nuclear Power Plant, Unit 1, "Integrated Leak Rate Test Unit 1 Containment."
28. Procedure STP M-662-2, Revision 7, Calvert Cliffs Nuclear Power Plant, Unit 2, "Integrated Leak Rate Test Unit 2 Containment."
29. Calculation No. C0-QU-002, Revision 1, Calvert Cliffs Nuclear Power Plant, Unit 2, "PRA Quantification (QU) Notebook," August 2010.
30. Calculation No. RSC 10-21, Revision 0, Calvert Cliffs Nuclear Power Plant, Unit 2, "Evaluation of Risk Significance of ILRT Extension," August 2010.
31. Armstrong, J., *Simplified Level 2 Modeling Guidelines: WOG PROJECT: PA-RMSC-0088*, Westinghouse, WCAP-16341-P, November 2005.
32. Calculation C0-LE-001, Revision 1, CENG, Units 1 and 2, "PRA Level 2 Notebook," May 2010.
33. Landale, J., PRAER No. C0-2010-012, CENG, C0-2010-012, August 2010.
34. Harrison, D., *Generic Component Fragilities for the GE Advanced BWR Seismic Analysis*, International Technology Corporation, September 1988.
35. Calculation No. RAN 97-031, IPEEE, Calvert Cliffs Nuclear Power Plant, "Individual Plant Examination of External Events," August 1997.
36. Report KLD-TR-21, *Development of Evacuation Time Estimates*, April 2008.
37. Letter L-14 -121, ML14111A291, FENOC Evaluation of the Proposed Amendment, Beaver Valley Power Station, Unit Nos. 1 and 2, April 2014.
38. Technical Letter Report ML112070867, *Containment Liner Corrosion Operating Experience Summary*, Revision 1, August 2011.

4.0 ASSUMPTIONS AND LIMITATIONS

The following assumptions were used in the calculation:

- The technical adequacy of the CCNPP PRA is consistent with the requirements of Regulatory Guide 1.200 as is relevant to this ILRT interval extension, as detailed in Attachment 1.
- The CCNPP Level 1 and Level 2 internal events PRA models provide representative results.
- It is appropriate to use the CCNPP internal events PRA model as a gauge to effectively describe the risk change attributable to the ILRT extension. An extensive sensitivity study is done in Section 5.3.1 to show the effect of including external event models for the ILRT extension. The IPEEE simplified seismic PRA [Reference 35] and the detailed Fire PRA (model 6.1M) are used for this sensitivity analysis. It is reasonable to assume that the impact from the ILRT extension (with respect to percent increases in population dose) will not substantially differ if detailed analysis of seismic events were to be included in the calculations.
- Accident classes describing radionuclide release end states are defined consistent with EPRI methodology [Reference 2].
- The representative containment leakage for Class 1 sequences is 1La. Class 3 accounts for increased leakage due to Type A inspection failures.
- The representative containment leakage for Class 3a sequences is 10La based on the previously approved methodology performed for Indian Point Unit 3 [Reference 8, Reference 9].
- The representative containment leakage for Class 3b sequences is 100La based on the guidance provided in EPRI Report No. 1009325, Revision 2-A (EPRI 1018243) [Reference 24].
- The Class 3b can be very conservatively categorized as LERF based on the previously approved methodology [Reference 8, Reference 9].
- The impact on population doses from containment bypass scenarios is not altered by the proposed ILRT extension, but is accounted for in the EPRI methodology as a separate entry for comparison purposes. Since the containment bypass contribution to population dose is fixed, no changes on the conclusions from this analysis will result from this separate categorization.
- The reduction in ILRT frequency does not impact the reliability of containment isolation valves to close in response to a containment isolation signal.

5.0 METHODOLOGY AND ANALYSIS

5.1 Inputs

This section summarizes the general resources available as input (Section 5.1.1) and the plant specific resources required (Section 5.1.2).

5.1.1 General Resources Available

Various industry studies on containment leakage risk assessment are briefly summarized here:

1. NUREG/CR-3539 [Reference 10]
2. NUREG/CR-4220 [Reference 11]
3. NUREG-1273 [Reference 12]
4. NUREG/CR-4330 [Reference 13]

5. EPRI TR-105189 [Reference 14]
6. NUREG-1493 [Reference 6]
7. EPRI TR-104285 [Reference 2]
8. NUREG-1150 [Reference 15] and NUREG/CR-4551 [Reference 7]
9. NEI Interim Guidance [Reference 3, Reference 20]
10. Calvert Cliffs liner corrosion analysis [Reference 5]
11. EPRI Report No. 1009325, Revision 2-A (EPRI 1018243), Appendix H [Reference 24]

This first study is applicable because it provides one basis for the threshold that could be used in the Level 2 PRA for the size of containment leakage that is considered significant and is to be included in the model. The second study is applicable because it provides a basis of the probability for significant pre-existing containment leakage at the time of a core damage accident. The third study is applicable because it is a subsequent study to NUREG/CR-4220 that undertook a more extensive evaluation of the same database. The fourth study provides an assessment of the impact of different containment leakage rates on plant risk. The fifth study provides an assessment of the impact on shutdown risk from ILRT test interval extension. The sixth study is the NRC's cost-benefit analysis of various alternative approaches regarding extending the test intervals and increasing the allowable leakage rates for containment integrated and local leak rate tests. The seventh study is an EPRI study of the impact of extending ILRT and LLRT test intervals on at-power public risk. The eighth study provides an ex-plant consequence analysis for a 50-mile radius surrounding a plant that is used as the basis for the consequence analysis of the ILRT interval extension for CCNPP. The ninth study includes the NEI recommended methodology (promulgated in two letters) for evaluating the risk associated with obtaining a one-time extension of the ILRT interval. The tenth study addresses the impact of age-related degradation of the containment liners on ILRT evaluations. Finally, the eleventh study builds on the previous work and includes a recommended methodology and template for evaluating the risk associated with a permanent 15-year extension of the ILRT interval.

NUREG/CR-3539 [Reference 10]

Oak Ridge National Laboratory documented a study of the impact of containment leak rates on public risk in NUREG/CR-3539. This study uses information from WASH-1400 [Reference 16] as the basis for its risk sensitivity calculations. ORNL concluded that the impact of leakage rates on LWR accident risks is relatively small.

NUREG/CR-4220 [Reference 11]

NUREG/CR-4220 is a study performed by Pacific Northwest Laboratories for the NRC in 1985. The study reviewed over two thousand LERs, ILRT reports and other related records to calculate the unavailability of containment due to leakage.

NUREG-1273 [Reference 12]

A subsequent NRC study, NUREG-1273, performed a more extensive evaluation of the NUREG/CR-4220 database. This assessment noted that about one-third of the reported events were leakages that were immediately detected and corrected. In addition, this study noted that local leak rate tests can detect "essentially all potential degradations" of the containment isolation system.

NUREG/CR-4330 [Reference 13]

NUREG/CR-4330 is a study that examined the risk impacts associated with increasing the allowable containment leakage rates. The details of this report have no direct impact on the

modeling approach of the ILRT test interval extension, as NUREG/CR-4330 focuses on leakage rate and the ILRT test interval extension study focuses on the frequency of testing intervals. However, the general conclusions of NUREG/CR-4330 are consistent with NUREG/CR-3539 and other similar containment leakage risk studies:

“...the effect of containment leakage on overall accident risk is small since risk is dominated by accident sequences that result in failure or bypass of containment.”

EPRI TR-105189 [Reference 14]

The EPRI study TR-105189 is useful to the ILRT test interval extension risk assessment because it provides insight regarding the impact of containment testing on shutdown risk. This study contains a quantitative evaluation (using the EPRI ORAM software) for two reference plants (a BWR-4 and a PWR) of the impact of extending ILRT and LLRT test intervals on shutdown risk. The conclusion from the study is that a small, but measurable, safety benefit is realized from extending the test intervals.

NUREG-1493 [Reference 6]

NUREG-1493 is the NRC's cost-benefit analysis for proposed alternatives to reduce containment leakage testing intervals and/or relax allowable leakage rates. The NRC conclusions are consistent with other similar containment leakage risk studies:

Reduction in ILRT frequency from 3 per 10 years to 1 per 20 years results in an “imperceptible” increase in risk.

Given the insensitivity of risk to the containment leak rate and the small fraction of leak paths detected solely by Type A testing, increasing the interval between integrated leak rate tests is possible with minimal impact on public risk.

EPRI TR-104285 [Reference 2]

Extending the risk assessment impact beyond shutdown (the earlier EPRI TR-105189 study), the EPRI TR-104285 study is a quantitative evaluation of the impact of extending ILRT and LLRT test intervals on at-power public risk. This study combined IPE Level 2 models with NUREG-1150 Level 3 population dose models to perform the analysis. The study also used the approach of NUREG-1493 in calculating the increase in pre-existing leakage probability due to extending the ILRT and LLRT test intervals.

EPRI TR-104285 uses a simplified Containment Event Tree to subdivide representative core damage frequencies into eight classes of containment response to a core damage accident:

1. Containment intact and isolated
2. Containment isolation failures dependent upon the core damage accident
3. Type A (ILRT) related containment isolation failures
4. Type B (LLRT) related containment isolation failures
5. Type C (LLRT) related containment isolation failures
6. Other penetration related containment isolation failures
7. Containment failures due to core damage accident phenomena
8. Containment bypass

Consistent with the other containment leakage risk assessment studies, this study concluded:

“...the proposed CLRT (Containment Leak Rate Tests) frequency changes would have a minimal safety impact. The change in risk determined by the analyses is small in both absolute and relative terms. For example, for the PWR analyzed, the change is about 0.04 person-rem per year...”

NUREG-1150 [Reference 15] and NUREG/CR-4551 [Reference 7]

NUREG-1150 and the technical basis, NUREG/CR-4551, provide an ex-plant consequence analysis for a spectrum of accidents including a severe accident with the containment remaining intact (i.e., Tech Spec Leakage). This ex-plant consequence analysis is calculated for the 50-mile radial area surrounding Surry. The ex-plant calculation can be delineated to total person-rem for each identified Accident Progression Bin (APB) from NUREG/CR-4551. With the CCNPP Level 2 model end-states assigned to one of the NUREG/CR-4551 APBs, it is considered adequate to represent CCNPP. (The meteorology and site differences other than population are assumed not to play a significant role in this evaluation.)

NEI Interim Guidance for Performing Risk Impact Assessments In Support of One-Time Extensions for Containment Integrated Leakage Rate Test Surveillance Intervals [Reference 3, Reference 20]

The guidance provided in this document builds on the EPRI risk impact assessment methodology [Reference 2] and the NRC performance-based containment leakage test program [Reference 6], and considers approaches utilized in various submittals, including Indian Point 3 (and associated NRC SER) and Crystal River.

Calvert Cliffs Response to Request for Additional Information Concerning the License Amendment for a One-Time Integrated Leakage Rate Test Extension [Reference 5]

This submittal to the NRC describes a method for determining the change in likelihood, due to extending the ILRT, of detecting liner corrosion, and the corresponding change in risk. The methodology was developed for Calvert Cliffs in response to a request for additional information regarding how the potential leakage due to age-related degradation mechanisms was factored into the risk assessment for the ILRT one-time extension. The Calvert Cliffs analysis was performed for a concrete cylinder and dome and a concrete base-mat, each with a steel liner.

EPRI Report No. 1009325, Revision 2-A, Risk Impact Assessment of Extended Integrated Leak Rate Testing Intervals [Reference 24]

This report provides a generally applicable assessment of the risk involved in extension of ILRT test intervals to permanent 15-year intervals. Appendix H of this document provides guidance for performing plant-specific supplemental risk impact assessments and builds on the previous EPRI risk impact assessment methodology [Reference 2] and the NRC performance-based containment leakage test program [Reference 6], and considers approaches utilized in various submittals, including Indian Point 3 (and associated NRC SER) and Crystal River.

The approach included in this guidance document is used in the CCNPP assessment to determine the estimated increase in risk associated with the ILRT extension. This document includes the bases for the values assigned in determining the probability of leakage for the EPRI Class 3a and 3b scenarios in this analysis, as described in Section 5.2.

5.1.2 Plant Specific Inputs

The plant-specific information used to perform the CCNPP ILRT Extension Risk Assessment includes the following:

- Level 1 Model results: Unit 1 [Reference 17] and Unit 2 [Reference 29]
- Level 2 Model results [Reference 17, Reference 18, Reference 19]
- Release category definitions used in the Level 2 Model [Reference 18, Reference 19]
- Dose within a 50-mile radius [Reference 19]

- ILRT results to demonstrate adequacy of the administrative and hardware issues [Reference 30]
- Containment failure probability data [Reference 18, References 32 and 33]

Level 1 Model

The Level 1 Internal Events PRA Model that is used for CCNPP is characteristic of the as-built plant. The current Level 1 model (CCNPP PRA Model Version 6.2a) [Reference 17] is a linked fault tree model, and was quantified with the total Internal Events Core Damage Frequency (CDF) = 1.61E-5/year for Unit 1 and CDF = 1.41E-5/year for Unit 2. The total External Event CDF (excluding seismic) = 3.24E-5/year for Unit 1 and 3.71E-5/year for Unit 2. Table 5-1 provides a summary of the Internal Events CDF results for CCNPP PRA Model Version 6.2a. Table 5-2 provides a summary of the External Events CDF results. The High Winds are included in CCNPP PRA Model Version 6.2a. The Fire PRA results come from Model Version 6.1M. The Seismic PRA results come from the IPEEE Seismic Analysis [Reference 35].

Table 5-1 – Internal Events CDF (CCNPP PRA Model Version 6.2a)

Internal Events	Unit 1 Frequency (per year)	Unit 2 Frequency (per year)
LOCAs	5.88E-6	7.70E-6
Internal Floods	6.18E-6	1.06E-6
Transients	3.40E-6	4.70E-6
ISLOCA	1.97E-7	1.97E-7
SGTR	4.71E-7	4.60E-7
Total Internal Events CDF	1.61E-5	1.41E-5
Total Internal Events CDF (Excluding ISLOCA & SGTR)	1.55E-5	1.34E-5

Table 5-2 – External Events CDF

External Events	Unit 1 Frequency (per year)	Unit 2 Frequency (per year)
Fire	3.15E-5	3.59E-5
High Winds	9.19E-7	1.23E-6
Seismic	1.29E-5	1.52E-5
Total External Events CDF	4.53E-5	5.23E-5

Note that the above Fire PRA values reflect the anticipated configuration of the plant upon full implementation of NFPA 805 and related plant modifications to resolve fire protection issues. Refer to Section 5.3.1.

Level 2 Model

The Level 2 Model that is used for CCNPP was developed with guidance from WCAP-16341-P to calculate the LERF contribution, as well as the other release end states evaluated in the model: INTACT, SERF (small early release frequency), and LATE [Reference 31]. The current LERF model (CCNPP PRA Model Version 6.2a) [Reference 17] is a linked fault tree model and was quantified with the total Unit 1 Internal Events LERF = 1.39E-6/year and Unit 2 Internal

Events LERF = $1.56E-6$ /year. The total Unit 1 External Event LERF (excluding seismic) = $2.99E-6$ /year and Unit 2 External Event LERF (excluding seismic) = $4.21E-6$ /year. Table 5-3 provides a summary of the Internal Events LERF results for CCNPP PRA Model Version 6.2a. Table 5-4 provides a summary of the External Events CDF results. The High Winds are included in CCNPP PRA Model Version 6.2a. The Fire PRA results come from Model Version 6.1M. The Seismic PRA results come from the IPEEE Seismic Analysis [Reference 35].

Table 5-3 – Internal Events LERF (CCNPP PRA Model Version 6.2a)

Internal Events	Unit 1 Frequency (per year)	Unit 2 Frequency (per year)
LOCAs	$3.26E-7$	$4.01E-7$
Internal Floods	$2.46E-7$	$2.17E-7$
Transients	$1.50E-7$	$2.84E-7$
ISLOCA	$1.97E-7$	$1.97E-7$
SGTR	$4.71E-7$	$4.60E-7$
Total Internal Events LERF	$1.39E-6$	$1.56E-6$

Table 5-4 – External Events LERF

External Events	Unit 1 Frequency (per year)	Unit 2 Frequency (per year)
Fire	$2.97E-6$	$4.17E-6$
High Winds	$2.21E-8$	$3.77E-8$
Seismic	$1.41E-6$	$1.66E-6$
Total External Events CDF	$4.40E-6$	$5.87E-6$

Note that the above Fire PRA values reflect the anticipated configuration of the plant upon full implementation of NFPA 805 and related plant modifications to resolve fire protection issues. Refer to Section 5.3.1.

Population Dose Calculations

The population dose calculation was performed for the CCNPP Severe Accident Mitigation Alternatives (SAMA) analyses [Reference 19] in 2010. Table 5-5 presents dose exposures calculated from methodology described in Reference 1 and data from Reference 19. Reference 19 provides the population dose (person-rem) for Classes 1, 2, 6, 7, and 8; Class 3a and 3b population dose values are calculated from the Class 1 population dose and represented as $10L_a$ and $100L_a$, respectively, as guidance in Reference 1 dictates.

Table 5-5 – Population Dose

Accident Class	Description	Release (person-rem)
1	Containment Remains Intact	$3.20E+04$
2	Containment Isolation Failures	$2.00E+07$
3a	Independent or Random Isolation Failures SMALL	$3.20E+05^1$
3b	Independent or Random Isolation Failures LARGE	$3.20E+06^2$

Accident Class	Description	Release (person-rem)
4	Isolation Failure in which pre-existing leakage is not dependent on sequence progression. Type B test Failures	n/a
5	Isolation Failure in which pre-existing leakage is not dependent on sequence progression. Type C test Failures	n/a
6	Isolation Failure that can be verified by IST/IS or surveillance	7.01E+06
7	Containment Failure induced by severe accident	5.61E+07
8	Accidents in which containment is by-passed	2.25E+07

1. $10 * L_a$
2. $100 * L_a$

Release Category Definitions

Table 5-6 defines the accident classes used in the ILRT extension evaluation, which is consistent with the EPRI methodology [Reference 2]. These containment failure classifications are used in this analysis to determine the risk impact of extending the Containment Type A test interval, as described in Section 5.2 of this report.

Class	Description
1	Containment remains intact including accident sequences that do not lead to containment failure in the long term. The release of fission products (and attendant consequences) is determined by the maximum allowable leakage rate values L_a , under Appendix J for that plant.
2	Containment isolation failures (as reported in the Individual Plant Examinations) including those accidents in which there is a failure to isolate the containment.
3	Independent (or random) isolation failures include those accidents in which the pre-existing isolation failure to seal (i.e., provide a leak-tight containment) is not dependent on the sequence in progress.
4	Independent (or random) isolation failures include those accidents in which the pre-existing isolation failure to seal is not dependent on the sequence in progress. This class is similar to Class 3 isolation failures, but is applicable to sequences involving Type B tests and their potential failures. These are the Type B-tested components that have isolated, but exhibit excessive leakage.
5	Independent (or random) isolation failures including those accidents in which the pre-existing isolation failure to seal is not dependent on the sequence in progress. This class is similar to Class 4 isolation failures, but is applicable to sequences involving Type C test and their potential failures.
6	Containment isolation failures including those leak paths covered in the plant test and maintenance requirements or verified per in-service inspection and testing (ISI/IST) program.
7	Accidents involving containment failure induced by severe accident phenomena. Changes in Appendix J testing requirements do not impact these accidents.
8	Accidents in which the containment is bypassed (either as an initial condition or induced by phenomena) are included in Class 8. Changes in Appendix J testing requirements do not impact these accidents.

5.1.3 Impact of Extension on Detection of Component Failures that Lead to Leakage (Small and Large)

The ILRT can detect a number of component failures such as liner breach, failure of certain bellows arrangements and failure of some sealing surfaces, which can lead to leakage. The proposed ILRT test interval extension may influence the conditional probability of detecting these types of failures. To ensure that this effect is properly addressed, the EPRI Class 3

accident class, as defined in Table 5-6, is divided into two sub-classes, Class 3a and Class 3b, representing small and large leakage failures respectively.

The probability of the EPRI Class 3a and Class 3b failures is determined consistent with the EPRI Guidance [Reference 24]. For Class 3a, the probability is based on the maximum likelihood estimate of failure (arithmetic average) from the available data (i.e., 2 “small” failures in 217 tests leads to “large” failures in 217 tests (i.e., $2/217 = 0.0092$). For Class 3b, the probability is based on the Jeffrey’s Non-Uniform Prior (i.e., $0.5/218 = 0.0023$).

In a follow-up letter [Reference 20] to their ILRT guidance document [Reference 3], NEI issued additional information concerning the potential that the calculated delta LERF values for several plants may fall above the “very small change” guidelines of the NRC Regulatory Guide 1.174 [Reference 4]. This additional NEI information includes a discussion of conservatisms in the quantitative guidance for Δ LERF. NEI describes ways to demonstrate that, using plant-specific calculations, the Δ LERF is smaller than that calculated by the simplified method.

The supplemental information states:

The methodology employed for determining LERF (Class 3b frequency) involves conservatively multiplying the CDF by the failure probability for this class (3b) of accident. This was done for simplicity and to maintain conservatism. However, some plant-specific accident classes leading to core damage are likely to include individual sequences that either may already (independently) cause a LERF or could never cause a LERF, and are thus not associated with a postulated large Type A containment leakage path (LERF). These contributors can be removed from Class 3b in the evaluation of LERF by multiplying the Class 3b probability by only that portion of CDF that may be impacted by Type A leakage.

The application of this additional guidance to the analysis for CCNPP, as detailed in Section 5.2, involves the following:

- The Class 2 and Class 8 sequences are subtracted from the CDF that is applied to Class 3b. To be consistent, the same change is made to the Class 3a CDF, even though these events are not considered LERF. Class 2 events refer to sequences with large pre-existing containment isolation failures; Class 8 events refer to sequences with containment bypass events. These sequences are already considered to contribute to LERF in the CCNPP Level 2 PRA analysis.
- A review of Class 1 accident sequences shows that several of these cases involve successful operation of containment sprays. For calculation of the Class 3b and Class 3a frequencies, the fraction of the Class 1 CDF associated with successful operation of containment sprays could also be subtracted. Successful operation of containment sprays result in lower containment pressure with subsequent reduction in containment leakage. This conservatism was removed for the CCNPP ILRT analysis, as detailed in Section 5.2.4.

Consistent with the NEI Guidance [Reference 3], the change in the leak detection probability can be estimated by comparing the average time that a leak could exist without detection. For example, the average time that a leak could go undetected with a three-year test interval is 1.5 years ($3 \text{ years} / 2$), and the average time that a leak could exist without detection for a ten-year interval is 5 years ($10 \text{ years} / 2$). This change would lead to a non-detection probability that is a factor of 3.33 ($5.0/1.5$) higher for the probability of a leak that is detectable only by ILRT testing. Correspondingly, an extension of the ILRT interval to 15 years can be estimated to lead to a factor of 5 ($(15/2)/1.5$) increase in the non-detection probability of a leak.

It should be noted that using the methodology discussed above is very conservative compared to previous submittals (e.g., the IP3 request for a one-time ILRT extension that was approved by the NRC [Reference 9]) because it does not factor in the possibility that the failures could be detected by other tests (e.g., the Type B local leak rate tests that will still occur). Eliminating this possibility conservatively over-estimates the factor increases attributable to the ILRT extension.

5.1.4 Impact of Extension on Detection of Steel Liner Corrosion that Leads to Leakage

An estimate of the likelihood and risk implications of corrosion-induced leakage of the steel liners occurring and going undetected during the extended test interval is evaluated using the methodology from the Calvert Cliffs liner corrosion analysis [Reference 5]. The Calvert Cliffs analysis was performed for a concrete cylinder and dome and a concrete base-mat, each with a steel liner.

The following approach is used to determine the change in likelihood, due to extending the ILRT, of detecting corrosion of the containment steel liner. This likelihood is then used to determine the resulting change in risk. Consistent with the Calvert Cliffs analysis, the following issues are addressed:

Differences between the containment base-mat and the containment cylinder and dome:

- The historical steel liner flaw likelihood due to concealed corrosion
- The impact of aging
- The corrosion leakage dependency on containment pressure
- The likelihood that visual inspections will be effective at detecting a flaw

Assumptions

- Consistent with the Calvert Cliffs analysis, a half failure is assumed for base-mat concealed liner corrosion due to the lack of identified failures. (See Table 5-7, Step 1)
- The two corrosion events used to estimate the liner flaw probability in the Calvert Cliffs previous analysis are assumed to still be applicable.
- Consistent with the Calvert Cliffs analysis, the estimated historical flaw probability is also limited to 5.5 years to reflect the years since September 1996 when 10 CFR 50.55a started requiring visual inspection. Additional success data was not used to limit the aging impact of this corrosion issue, even though inspections were being performed prior to this date (and have been performed since the time frame of the Calvert Cliffs analysis), and there is no evidence that additional corrosion issues were identified (See Table 5-7, Step 1).
- Consistent with the Calvert Cliffs analysis, the steel liner flaw likelihood is assumed to double every five years. This is based solely on judgment and is included in this analysis to address the increased likelihood of corrosion as the steel liner ages (See Table 5-7, Steps 2 and 3). Sensitivity studies are included that address doubling this rate every ten years and every two years.
- In the Calvert Cliffs analysis, the likelihood of the containment atmosphere reaching the outside atmosphere, given that a liner flaw exists, was estimated as 1.1% for the cylinder and dome, and 0.11% (10% of the cylinder failure probability) for the base-mat. These values were determined from an assessment of the probability versus containment pressure. For CCNPP, the ILRT maximum pressure is psig 50 [References 27 and 28] and ultimate pressure of 132 psig [References 32 and 33]. Probabilities of 1% for the cylinder and dome, and 0.1% for the base-mat are used in this analysis, and sensitivity studies are included in Section 5.3.2 (See Table 5-7, Step 4).
- Consistent with the Calvert Cliffs analysis, the likelihood of leakage escape (due to crack

formation) in the base-mat region is considered to be less likely than the containment cylinder and dome region (See Table 5-7, Step 4).

- Consistent with the Calvert Cliffs analysis, a 5% visual inspection detection failure likelihood given the flaw is visible and a total detection failure likelihood of 10% is used. To date, all liner corrosion events have been detected through visual inspection (See Table 5-7, Step 5).

Consistent with the Calvert Cliffs analysis, all non-detectable containment failures are assumed to result in early releases. This approach avoids a detailed analysis of containment failure timing and operator recovery actions.

Table 5-7 – Steel Liner Corrosion Base Case

Step	Description	Containment Cylinder and Dome (85%)	Containment Basemat (15%)
1	Historical liner flaw likelihood	Events: 2	Events: 0
	Failure data: containment location specific	(Brunswick 2 and North Anna 2) $2 / (70 \times 5.5) = 5.19E-03$	Assume a half failure $0.5 / (70 \times 5.5) = 1.30E-03$
	Success data: based on 70 steel-lined containments and 5.5 years since the 10CFR 50.55a requirements of periodic visual inspections of containment surfaces		
2	Aged adjusted liner flaw likelihood During the 15-year interval, assume failure rate doubles every five years (14.9% increase per year). The average for the 5th to 10th year set to the historical failure rate.	Year	Year
		1	1
		average 5-10	average 5-10
		15	15
		Failure rate	Failure rate
		2.05E-03	5.13E-04
		5.19E-03	1.30E-03
		1.43E-02	3.57E-03
		15 year average = 6.44E-03	15 year average = 1.61E-03
3	Increase in flaw likelihood between 3 and 15 years Uses aged adjusted liner flaw likelihood (Step 2), assuming failure rate doubles every five years.	0.73% (1 to 3 years)	0.18% (1 to 3 years)
		4.18% (1 to 10 years)	1.04% (1 to 10 years)
		9.66% (1 to 15 years)	2.41% (1 to 15 years)
4	Likelihood of breach in containment given liner flaw	1%	0.1%
5	Visual inspection detection failure likelihood	10%	100%
		5% failure to identify visual flaws plus 5% likelihood that the flaw is not visible (not through-cylinder but could be detected by ILRT). All events have been detected through visual inspection. 5% visible failure detection is a conservative assumption.	Cannot be visually inspected
6	Likelihood of non-detected containment leakage (Steps 3 x 4 x 5)	0.00073% (3 years)	0.000180% (3 years)
		$0.73\% \times 1\% \times 10\%$	$0.18\% \times 0.1\% \times 100\%$
		0.00418% (10 years)	0.00104% (10 years)
		$4.18\% \times 1\% \times 10\%$	$1.04\% \times 0.1\% \times 100\%$
		0.00966% (15 years)	0.00241% (15 years)
		$9.66\% \times 1\% \times 10\%$	$2.41\% \times 0.1\% \times 100\%$

The total likelihood of the corrosion-induced, non-detected containment leakage is the sum of Step 6 for the containment cylinder and dome, and the containment base-mat, as summarized below for CCNPP.

Table 5-8 – Total Likelihood on Non-Detected Containment Leakage Due to Corrosion for CCNPP

Description
At 3 years: $0.00073\% + 0.000180\% = 0.00091\%$
At 10 years: $0.00418\% + 0.00104\% = 0.00522\%$
At 15 years: $0.00966\% + 0.00241\% = 0.01207\%$

The above factors are applied to those core damage accidents that are not already independently LERF or that could never result in LERF.

The two corrosion events that were initiated from the non-visible (backside) portion of the containment liner used to estimate the liner flaw probability in the Calvert Cliffs analysis are assumed to be applicable to this containment analysis. These events, one at North Anna Unit 2 (September 1999) caused by a timber embedded in the concrete immediately behind the containment liner, and one at Brunswick Unit 2 (April 1999) caused by a cloth work glove embedded in the concrete next to the liner, were initiated from the nonvisible (backside) portion of the containment liner. A search of the NRG website LER database identified two additional events have occurred since the Calvert Cliffs analysis was performed. In January 2000, a 3/16-inch circular through-liner hole was found at Cook Nuclear Plant Unit 2 caused by a wooden brush handle embedded immediately behind the containment liner. The other event occurred in April 2009, where a through-liner hole approximately 3/8-inch by 1-inch in size was identified in the Beaver Valley Power Station Unit 1 (BVPS-1) containment liner caused by pitting originating from the concrete side due to a piece of wood that was left behind during the original construction that came in contact with the steel liner. Two other containment liner through wall hole events occurred at Turkey Point Units 3 and 4 in October 2010 and November 2006, respectively. However, these events originated from the visible side caused by the failure of the coating system, which was not designed for periodic immersion service, and are not considered to be applicable to this analysis. More recently, in October 2013, some through-wall containment liner holes were identified at BVPS-1, with a combined total area of approximately 0.395 square inches. The cause of these through wall liner holes was attributed to corrosion originating from the outside concrete surface due to the presence of rayon fiber foreign material that was left behind during the original construction and was contacting the steel liner. For risk evaluation purposes, these five total corrosion events occurring in 66 operating plants with steel containment liners over a 17.1 year period from September 1996 to October 4, 2013 (i.e., $5/(66 \cdot 17.1) = 4.43E-03$) are bounded by the estimated historical flaw probability based on the two events in the 5.5 year period of the Calvert Cliffs analysis (i.e., $2/(70 \cdot 5.5) = 5.19E-03$) incorporated in the EPRI guidance.

5.2 Analysis

The application of the approach based on the guidance contained in EPRI Report No. 1009325, Revision 2-A, Appendix H [Reference 24], EPRI TR-104285 [Reference 2] and previous risk assessment submittals on this subject [References 5, 8, 21, 22, and 23] have led to the following results. The results are displayed according to the eight accident classes defined in the EPRI report, as described in Table 5-6.

The analysis performed examined CCNPP-specific accident sequences in which the containment remains intact or the containment is impaired. Specifically, the breakdown of the severe accidents, contributing to risk, was considered in the following manner:

- Core damage sequences in which the containment remains intact initially and in the long term (EPRI TR-104285, Class 1 sequences [Reference 2]).
- Core damage sequences in which containment integrity is impaired due to random isolation failures of plant components other than those associated with Type B or Type C test components. For example, liner breach or bellow leakage (EPRI TR-104285, Class 3 sequences [Reference 2]).
- Accident sequences involving containment bypassed (EPRI TR-104285, Class 8 sequences [Reference 2]), large containment isolation failures (EPRI TR-104285, Class 2 sequences [Reference 2]), and small containment isolation “failure-to-seal” events (EPRI TR-104285, Class 4 and 5 sequences [Reference 2]) are accounted for in this evaluation as part of the baseline risk profile. However, they are not affected by the ILRT frequency change.
- Class 4 and 5 sequences are impacted by changes in Type B and C test intervals; therefore, changes in the Type A test interval do not impact these sequences.

Table 5-9 – EPRI Accident Class Definitions

Accident Classes (Containment Release Type)	Description
1	No Containment Failure
2	Large Isolation Failures (Failure to Close)
3a	Small Isolation Failures (Liner Breach)
3b	Large Isolation Failures (Liner Breach)
4	Small Isolation Failures (Failure to Seal – Type B)
5	Small Isolation Failures (Failure to Seal – Type C)
6	Other Isolation Failures (e.g., Dependent Failures)
7	Failures Induced by Phenomena (Early and Late)
8	Bypass (Interfacing System LOCA)
CDF	All CET End States (Including Very Low and No Release)

The steps taken to perform this risk assessment evaluation are as follows:

Step 1 - Quantify the baseline risk in terms of frequency per reactor year for each of the accident classes presented in Table 5-9.

Step 2 - Develop plant-specific person-rem dose (population dose) per reactor year for each of the eight accident classes.

Step 3 - Evaluate risk impact of extending Type A test interval from 3 in 10 years to 1 in 15 years and 1 in 10 years to 1 in 15 years.

Step 4 - Determine the change in risk in terms of Large Early Release Frequency (LERF) in accordance with RG 1.174 [Reference 4].

Step 5 - Determine the impact on the Conditional Containment Failure Probability (CCFP).

5.2.1 Step 1 – Quantify the Baseline Risk in Terms of Frequency per Reactor Year

As previously described, the extension of the Type A interval does not influence those accident progressions that involve large containment isolation failures, Type B or Type C testing, or

containment failure induced by severe accident phenomena.

For the assessment of ILRT impacts on the risk profile, the potential for pre-existing leaks is included in the model. (These events are represented by the Class 3 sequences in EPRI TR-104285 [Reference 2].) The question on containment integrity was modified to include the probability of a liner breach or bellows failure (due to excessive leakage) at the time of core damage. Two failure modes were considered for the Class 3 sequences. These are Class 3a (small breach) and Class 3b (large breach).

The frequencies for the severe accident classes defined in Table 5-9 were developed for CCNPP by first determining the frequencies for Classes 1, 2, 6, 7, and 8. Table 5-10 provides a correlation of the adjusted release category frequencies and the EPRI release classes in Table 5-9. Table 5-10 provides the CCNPP-specific frequencies for each Level 2 release category. Table 5-11 presents the grouping of each endstate in EPRI Classes based on the associated description. Table 5-12 presents the LERF sequence description, frequency and EPRI category for each sequence and the totals of each EPRI classification. Table 5-13 provides a summary of the accident sequence frequencies that can lead to radionuclide release to the public and have been derived consistent with the definitions of accident classes defined in EPRI TR-104285 [Reference 2], the NEI Interim Guidance [Reference 3], and guidance provided in EPRI Report No. 1009325, Revision 2-A [Reference 24]. Adjustments were made to the Class 3b and hence Class 1 frequencies to account for the impact of undetected corrosion of the steel liner per the methodology described in Section 5.1.4. Note: calculations were performed with more digits than shown in this section. Therefore, minor differences may occur if the calculations in this section are followed explicitly.

Class 3 Sequences. This group consists of all core damage accident progression bins for which a pre-existing leakage in the containment structure (e.g., containment liner) exists that can only be detected by performing a Type A ILRT. The probability of leakage detectable by a Type A ILRT is calculated to determine the impact of extending the testing interval. The Class 3 calculation is divided into two classes: Class 3a is defined as a small liner breach ($L_a < \text{leakage} < 10L_a$), and Class 3b is defined as a large liner breach ($10L_a < \text{leakage} < 100L_a$).

Data reported in EPRI 1009325, Revision 2-A [Reference 24] states that two events could have been detected only during the performance of an ILRT and thus impact risk due to change in ILRT frequency. There were a total of 217 successful ILRTs during this data collection period. Therefore, the probability of leakage is determined for Class 3a as shown in the following equation:

$$P_{class3a} = \frac{2}{217} = 0.0092$$

Multiplying the CDF by the probability of a Class 3a leak yields the Class 3a frequency contribution in accordance with guidance provided in Reference 24. As described in Section 5.1.3, additional consideration is made to not apply failure probabilities on those cases that are already LERF scenarios (i.e., the Class 2 and Class 8 contributions). Therefore, these LERF contributions from CDF are removed. Therefore, the frequency of a Class 3a failure is calculated by the following equation:

$$Freq_{U1class3a} = P_{class3a} * (CDF_{U1} - Class2_{U1} - Class8_{U1}) = \frac{2}{217} * (1.61E-5 - 5.01E-8 - 6.77E-7)$$

$$Freq_{U1class3a} = 1.42E-7$$

$$Freq_{U2class3a} = P_{class3a} * (CDF_{U2} - Class2_{U2} - Class8_{U2}) = \frac{2}{217} * (1.41E-5 - 4.29E-8 - 6.72E-7)$$

$$Freq_{U2class3a} = 1.23E-7$$

In the database of 217 ILRTs, there are zero containment leakage events that could result in a large early release. Therefore, the Jeffreys Non-Informed Prior is used to estimate a failure rate and is illustrated in the following equations:

$$\text{Jeffreys Failure Probability} = \frac{\text{Number of Failures} + 1/2}{\text{Number of Tests} + 1}$$

$$P_{\text{class3b}} = \frac{0 + 1/2}{217 + 1} = 0.0023$$

The frequency of a Class 3b failure is calculated by the following equation:

$$\text{Freq}_{U1\text{class3b}} = P_{\text{class3b}} * (\text{CDF}_{U1} - \text{Class2}_{U1} - \text{Class8}_{U1}) = \frac{.5}{218} * (1.61\text{E-}5 - 5.01\text{E-}8 - 6.77\text{E-}7)$$

$$\text{Freq}_{U1\text{class3b}} = 3.52\text{E-}8$$

$$\text{Freq}_{U2\text{class3b}} = P_{\text{class3b}} * (\text{CDF}_{U2} - \text{Class2}_{U2} - \text{Class8}_{U2}) = \frac{.5}{218} * (1.41\text{E-}5 - 4.34\text{E-}8 - 6.72\text{E-}7)$$

$$\text{Freq}_{U2\text{class3b}} = 3.07\text{E-}8$$

For this analysis, the associated containment leakage for Class 3a is 10L_a and for Class 3b is 100L_a. These assignments are consistent with the guidance provided in Reference 24.

Class 1 Sequences. This group consists of all core damage accident progression bins for which the containment remains intact (modeled as Technical Specification Leakage). The frequency per year is initially determined from the EPRI Accident Class 1 frequency listed in Table 5-12 and then subtracting the EPRI Class 3a and 3b frequency (to preserve total CDF), calculated below:

$$\text{Freq}_{U1\text{class1}} = \text{Freq}_{U1\text{class1}} - (\text{Freq}_{U1\text{class3a}} - \text{Freq}_{U1\text{class3b}})$$

$$\text{Freq}_{U2\text{class1}} = \text{Freq}_{U2\text{class1}} - (\text{Freq}_{U2\text{class3a}} - \text{Freq}_{U2\text{class3b}})$$

Class 2 Sequences. This group consists of core damage accident progression bins with large containment isolation failures. The frequency per year for these sequences is obtained from the EPRI Accident Class 2 frequency listed in Table 5-12.

Class 4 Sequences. This group consists of all core damage accident progression bins for which containment isolation failure-to-seal of Type B test components occurs. Because these failures are detected by Type B tests which are unaffected by the Type A ILRT, this group is not evaluated any further in the analysis, consistent with approved methodology.

Class 5 Sequences. This group consists of all core damage accident progression bins for which a containment isolation failure-to-seal of Type C test components. Because the failures are detected by Type C tests which are unaffected by the Type A ILRT, this group is not evaluated any further in this analysis, consistent with approved methodology.

Class 6 Sequences. These are sequences that involve core damage accident progression bins for which a failure-to-seal containment leakage due to failure to isolate the containment occurs. These sequences are dominated by misalignment of containment isolation valves following a test/maintenance evolution. For CCNPP, this class is defined as the SERF category. The frequency per year for these sequences is obtained from the EPRI Accident Class 6 frequency listed in Table 5-12.

Class 7 Sequences. This group consists of all core damage accident progression bins in which containment failure induced by severe accident phenomena occurs (e.g., overpressure). For this analysis, the frequency is determined from the EPRI Accident Class 7 frequency listed in Table 5-12.

Class 8 Sequences. This group consists of all core damage accident progression bins in which containment bypass or SGTR occurs. For this analysis, the frequency is determined from the EPRI Accident Class 8 frequency listed in Table 5-12.

Release Category	EPRI Category	Unit 1 Frequency (/yr)	Unit 2 Frequency (/yr)
INTACT	Class 1	6.76E-06	5.12E-06
LERF	Classes 2,7, 8	1.39E-06	1.56E-06
SERF	Class 6	1.87E-06	1.25E-06
LATE	Class 1	6.06E-06	6.17E-06 ¹
Total (CDF)	N/A	1.61E-05	1.41E-05

1. Unit 2 LATE was quantified at 5E-12 truncation. The other end states were quantified at 1E-12.

LERF quantification is distributed into EPRI categories based on end states. Table 5-11 shows this distribution.

CCNPP LERF End State	Description of Outcome	EPRI Category	Unit 1 Frequency (/yr)	Unit 2 Frequency (/yr)
LERF01	Containment failure following high-pressure (HP) vessel breach (VB)	7	3.66E-07	3.80E-07
LERF02	Containment failure following HP VB	7	4.87E-10	5.45E-08
LERF03	Containment failure following low pressure (LP) VB	7	4.25E-11	1.89E-07
LERF04	Temperature induced (TI) SGTR	8	0.00E+00	3.17E-09
LERF05	Containment failure following LP VB	7	5.04E-08	9.61E-08
LERF06	Pressure induced (PI) SGTR	8	0.00E+00	0.00E+00
LERF07	Containment failure following LP VB	7	1.09E-08	9.68E-09
LERF08	Loss of isolation	2	3.34E-08	3.72E-08
LERF09	Containment bypass	8	6.68E-07	6.56E-07
LERF10	Containment failure following LP VB	7	1.37E-07	5.59E-08
LERF11	Containment failure following HP VB	7	2.01E-08	1.40E-08
LERF12	Containment failure following LP VB	7	5.14E-08	2.85E-08
LERF13	TI-SGTR	8	8.75E-09	1.21E-08
LERF14	Containment failure following LP VB	7	1.27E-08	7.63E-09
LERF15	PI-SGTR	8	0.00E+00	0.00E+00
LERF16	Containment failure following LP VB	7	0.00E+00	0.00E+00
LERF17	Loss of isolation	2	3.05E-08	1.71E-08
LERF18	Containment bypass	8	4.94E-10	5.23E-10
Contribution to EPRI Classification 2			6.39E-08	5.43E-08
Contribution to EPRI Classification 7			6.49E-07	8.35E-07
Contribution to EPRI Classification 8			6.77E-07	6.72E-07
Total LERF			1.39E-06	1.56E-06

Table 5-12 – Release Category Frequencies

Release Category	EPRI Category	Unit 1 Frequency (/yr)	Unit 2 Frequency (/yr)
INTACT + LATE ¹	Class 1	1.28E-05 ⁶	1.13E-05 ⁶
LERF ²	Class 2	5.01E-08 ⁶	4.34E-08 ⁶
SERF ³	Class 6	1.87E-06	1.25E-06
LERF ⁴	Class 7	6.49E-07	8.35E-07
LERF ⁵	Class 8	6.77E-07	6.72E-07
Total (CDF)		1.61E-5	1.41E-5

1. The EPRI Class 1 category consists of INTACT and LATE failures. A LATE failure is classified as intact due to the long time until failure and is consistent with guidance in Reference 24.
2. The EPRI Class 2 category consists of CCNPP assigned LERF contribution associated with isolation failures as re-categorized in Table 5-11 with pre-event containment liner failure removed (see note 6).
3. The EPRI Class 6 category consists of CCNPP assigned scrubbed isolation failures in SERF.
4. The EPRI Class 7 category consists of the CCNPP assigned LERF contribution associated with phenomenological failures as re-categorized in Table 5-11.
5. The EPRI Class 8 category consists of the CCNPP assigned LERF contribution associated with bypass or SGTR failures as re-categorized in Table 5-11.
6. The level 2 model contains a bounding contribution associated with pre-event containment liner failure. To preclude influencing the current detailed assessment, the contribution associated with this failure is adjusted by removal of the bounding estimate from Class 2 and adding it to the intact containment case (Class 1). The Unit 1 pre-event containment liner failure value is 1.385E-8; the Unit 2 value is 1.094E-8. These values are the LERF contributions from events FAIL_LEAK and FAIL_LEAK_2 for Units 1 and 2, respectively.

Table 5-13 – Baseline Risk Profile

Class	Description	Unit 1 Frequency (/yr)	Unit 2 Frequency (/yr)
1	No containment failure	1.27E-05 ²	1.12E-05 ²
2	Large containment isolation failures	5.01E-08	4.34E-08
3a	Small isolation failures (liner breach)	1.42E-07	1.23E-07
3b	Large isolation failures (liner breach)	3.52E-08	3.07E-08
4	Small isolation failures - failure to seal (type B)	ϵ^1	ϵ^1
5	Small isolation failures - failure to seal (type C)	ϵ^1	ϵ^1
6	Containment isolation failures (dependent failure, personnel errors)	1.87E-06	1.25E-06
7	Severe accident phenomena induced failure (early and late)	6.49E-07	8.35E-07
8	Containment bypass	6.77E-07	6.72E-07
Total		1.61E-05	1.41E-05

1. ϵ represents a probabilistically insignificant value.
2. The Class 3a and 3b frequencies are subtracted from Class 1 to preserve total CDF.

5.2.2 Step 2 – Develop Plant-Specific Person-Rem Dose (Population Dose)

Plant-specific release analyses were performed to estimate the person-rem doses to the population within a 50-mile radius from the plant. The releases are based on CCNPP-specific dose calculations summarized on Table 5-5. Table 5-14 provides a correlation of CCNPP population dose to EPRI Accident Class. Table 5-15 provides population dose for each EPRI accident class.

The population dose for EPRI Accident Classes 3a and 3b were calculated based on the guidance provided in EPRI Report No. 1009325, Revision 2-A [Reference 24] as follows:

$$\text{EPRI Class 3a Population Dose} = 10 * 3.20E+4 = 3.20E+5$$

$$\text{EPRI Class 3b Population Dose} = 100 * 3.20E+4 = 3.20E+6$$

Table 5-14 – Mapping of Population Dose to EPRI Accident Class

Release Category	EPRI Category	Unit 1 Frequency (/yr)	Unit 1 Dose (person-rem)	Unit 2 Frequency (/yr)	Unit 2 Dose (person-rem)
INTACT + LATE	Class 1	1.28E-05	3.20E+04	1.13E-05	3.20E+04
LERF	Class 2	5.01E-08	2.00E+07	4.34E-08	2.00E+07
SERF	Class 6	1.87E-06	7.01E+06	1.25E-06	7.01E+06
LERF	Class 7	6.49E-07	5.61E+07	8.35E-07	5.61E+07
LERF	Class 8	6.77E-07	2.25E+07	6.72E-07	2.25E+07

Table 5-15 – Baseline Population Doses

Class	Description	Population Dose (person-rem)
1	No containment failure	3.20E+04
2	Large containment isolation failures	2.00E+07
3a	Small isolation failures (liner breach)	3.20E+05 ¹
3b	Large isolation failures (liner breach)	3.20E+06 ²
4	Small isolation failures - failure to seal (type B)	N/A
5	Small isolation failures - failure to seal (type C)	N/A
6	Containment isolation failures (dependent failure, personnel errors)	7.01E+06
7	Severe accident phenomena induced failure (early and late)	5.61E+07
8	Containment bypass	2.25E+07

1. $10 * L_a$
2. $100 * L_a$

5.2.3 Step 3 – Evaluate Risk Impact of Extending Type A Test Interval from 10 to 15 Years

The next step is to evaluate the risk impact of extending the test interval from its current 10-year interval to a 15-year interval. To do this, an evaluation must first be made of the risk associated with the 10-year interval, since the base case applies to 3-year interval (i.e., a simplified representation of a 3-to-10 interval).

Risk Impact Due to 10-Year Test Interval

As previously stated, Type A tests impact only Class 3 sequences. For Class 3 sequences, the release magnitude is not impacted by the change in test interval (a small or large breach remains the same, even though the probability of not detecting the breach increases). Thus, only the frequency of Class 3a and Class 3b sequences is impacted. The risk contribution is

changed based on the NEI guidance as described in Section 5.1.3 by a factor of 10/3 compared to the base case values. The Class 3a and 3b frequencies are calculated as follows:

$$Freq_{U1Class3a10yr} = \frac{10}{3} * \frac{2}{217} * (CDF_{U1} - Class2_{U1} - Class8_{U1}) = \frac{10}{3} * \frac{2}{217} * 1.54E-5 = 4.72E-7$$

$$Freq_{U2Class3a10yr} = \frac{10}{3} * \frac{2}{217} * (CDF_{U2} - Class2_{U2} - Class8_{U2}) = \frac{10}{3} * \frac{2}{217} * 1.34E-5 = 4.11E-7$$

$$Freq_{U1Class3b10yr} = \frac{10}{3} * \frac{.5}{218} * (CDF_{U1} - Class2_{U1} - Class8_{U1}) = \frac{10}{3} * \frac{.5}{218} * 1.54E-5 = 1.17E-7$$

$$Freq_{U2Class3b10yr} = \frac{10}{3} * \frac{.5}{218} * (CDF_{U2} - Class2_{U2} - Class8_{U2}) = \frac{10}{3} * \frac{.5}{218} * 1.34E-5 = 1.02E-7$$

The results of the calculation for a 10-year interval for Units 1 and 2 interval are presented in Tables 5-16 and 5-17, respectively.

Risk Impact Due to 15-Year Test Interval

The risk contribution for a 15-year interval is calculated in a manner similar to the 10-year interval. The difference is in the increase in probability of leakage in Classes 3a and 3b. For this case, the value used in the analysis is a factor of 5 compared to the 3-year interval value, as described in Section 5.1.3. The Class 3a and 3b frequencies are calculated as follows:

$$Freq_{U1Class3a15yr} = \frac{15}{3} * \frac{2}{217} * (CDF_{U1} - Class2_{U1} - Class8_{U1}) = 5 * \frac{2}{217} * 1.54E-5 = 7.08E-7$$

$$Freq_{U2Class3a15yr} = \frac{15}{3} * \frac{2}{217} * (CDF_{U2} - Class2_{U2} - Class8_{U2}) = 5 * \frac{2}{217} * 1.34E-5 = 6.17E-7$$

$$Freq_{U1Class3b15yr} = \frac{15}{3} * \frac{.5}{218} * (CDF_{U1} - Class2_{U1} - Class8_{U1}) = 5 * \frac{.5}{218} * 1.54E-5 = 1.76E-7$$

$$Freq_{U2Class3b15yr} = \frac{15}{3} * \frac{.5}{218} * (CDF_{U2} - Class2_{U2} - Class8_{U2}) = 5 * \frac{.5}{218} * 1.34E-5 = 1.53E-7$$

The results of the calculation for a 15-year interval for Units 1 and 2 are presented in Table 5-18 and 5-19.

Table 5-16 – Unit 1 Risk Profile for Once in 10 Year ILRT

Class	Description	Frequency (/yr)	Contribution (%)	Population Dose (person-rem)	Population Dose Rate (person-rem/yr)
1	No containment failure ¹	1.23E-05	76.17%	3.20E+04	3.92E-01
2	Large containment isolation failures	5.01E-08	0.31%	2.00E+07	1.00E+00
3a	Small isolation failures (liner breach)	4.72E-07	2.93%	3.20E+05	1.51E-01
3b	Large isolation failures (liner breach)	1.17E-07	0.73%	3.20E+06	3.76E-01
4	Small isolation failures - failure to seal (type B)	ε ¹	ε ¹	ε ¹	ε ¹
5	Small isolation failures - failure to seal (type C)	ε ¹	ε ¹	ε ¹	ε ¹
6	Containment isolation failures (dependent failure, personnel errors)	1.87E-06	11.61%	7.01E+06	1.31E+01

Table 5-16 – Unit 1 Risk Profile for Once in 10 Year ILRT

Class	Description	Frequency (/yr)	Contribution (%)	Population Dose (person-rem)	Population Dose Rate (person-rem/yr)
7	Severe accident phenomena induced failure (early and late)	6.49E-07	4.04%	5.61E+07	3.64E+01
8	Containment bypass	6.77E-07	4.21%	2.25E+07	1.52E+01
Total		1.61E-05			6.67E+01

1. ϵ represents a probabilistically insignificant value.
2. The Class 1 frequency is reduced by the frequency of Class 3a and Class 3b in order to preserve total CDF.

Table 5-17 – Unit 2 Risk Profile for Once in 10 Year ILRT

Class	Description	Frequency (/yr)	Contribution (%)	Population Dose (person-rem)	Population Dose Rate (person-rem/yr)
1	No containment failure ¹	1.08E-05	76.51%	3.20E+04	3.45E-01
2	Large containment isolation failures	4.34E-08	0.31%	2.00E+07	8.67E-01
3a	Small isolation failures (liner breach)	4.11E-07	2.92%	3.20E+05	1.32E-01
3b	Large isolation failures (liner breach)	1.02E-07	0.73%	3.20E+06	3.28E-01
4	Small isolation failures - failure to seal (type B)	ϵ	ϵ	ϵ	ϵ
5	Small isolation failures - failure to seal (type C)	ϵ	ϵ	ϵ	ϵ
6	Containment isolation failures (dependent failure, personnel errors)	1.25E-06	8.85%	7.01E+06	8.75E+00
7	Severe accident phenomena induced failure (early and late)	8.35E-07	5.92%	5.61E+07	4.69E+01
8	Containment bypass	6.72E-07	4.76%	2.25E+07	1.51E+01
Total		1.41E-05			7.24E+01

1. The Class 1 frequency is reduced by the frequency of Class 3a and Class 3b in order to preserve total CDF.

Table 5-18 – Unit 1 Risk Profile for Once in 15 Year ILRT

Class	Description	Frequency (/yr)	Contribution (%)	Population Dose (person-rem)	Population Dose Rate (person-rem/yr)
1	No containment failure ¹	1.20E-05	74.34%	3.20E+04	3.83E-01
2	Large containment isolation failures	5.01E-08	0.31%	2.00E+07	1.00E+00
3a	Small isolation failures (liner breach)	7.08E-07	4.40%	3.20E+05	2.26E-01
3b	Large isolation failures (liner breach)	1.76E-07	1.09%	3.20E+06	5.64E-01

Table 5-18 – Unit 1 Risk Profile for Once in 15 Year ILRT

Class	Description	Frequency (/yr)	Contribution (%)	Population Dose (person-rem)	Population Dose Rate (person-rem/yr)
4	Small isolation failures - failure to seal (type B)	ϵ^1	ϵ^1	ϵ^1	ϵ^1
5	Small isolation failures - failure to seal (type C)	ϵ^1	ϵ^1	ϵ^1	ϵ^1
6	Containment isolation failures (dependent failure, personnel errors)	1.87E-06	11.61%	7.01E+06	1.31E+01
7	Severe accident phenomena induced failure (early and late)	6.49E-07	4.04%	5.61E+07	3.64E+01
8	Containment bypass	6.77E-07	4.21%	2.25E+07	1.52E+01
Total		1.61E-05			6.69E+01

1. The Class 1 frequency is reduced by the frequency of Class 3a and Class 3b in order to preserve total CDF.

Table 5-19 – Unit 2 Risk Profile for Once in 15 Year ILRT

Class	Description	Frequency (/yr)	Contribution (%)	Population Dose (person-rem)	Population Dose Rate (person-rem/yr)
1	No containment failure ¹	1.05E-05	74.69%	3.20E+04	3.37E-01
2	Large containment isolation failures	4.34E-08	0.31%	2.00E+07	8.67E-01
3a	Small isolation failures (liner breach)	6.17E-07	4.37%	3.20E+05	1.97E-01
3b	Large isolation failures (liner breach)	1.54E-07	1.09%	3.20E+06	4.91E-01
4	Small isolation failures - failure to seal (type B)	ϵ^1	ϵ^1	ϵ^1	ϵ^1
5	Small isolation failures - failure to seal (type C)	ϵ^1	ϵ^1	ϵ^1	ϵ^1
6	Containment isolation failures (dependent failure, personnel errors)	1.25E-06	8.85%	7.01E+06	8.75E+00
7	Severe accident phenomena induced failure (early and late)	8.35E-07	5.92%	5.61E+07	4.69E+01
8	Containment bypass	6.72E-07	4.76%	2.25E+07	1.51E+01
Total		1.41E-05			7.26E+01

1. The Class 1 frequency is reduced by the frequency of Class 3a and Class 3b in order to preserve total CDF.

5.2.4 Step 4 – Determine the Change in Risk in Terms of LERF

The risk increase associated with extending the ILRT interval involves the potential that a core damage event that normally would result in only a small radioactive release from an intact containment could, in fact, result in a larger release due to the increase in probability of failure to

detect a pre-existing leak. With strict adherence to the EPRI guidance, 100% of the Class 3b contribution would be considered LERF.

Regulatory Guide 1.174 [Reference 4] provides guidance for determining the risk impact of plant-specific changes to the licensing basis. RG 1.174 [Reference 4] defines very small changes in risk as resulting in increases of CDF less than 10^{-6} /year and increases in LERF less than 10^{-7} /year, and small changes in LERF as less than 10^{-6} /year. Since containment overpressure is not required in support of ECCS performance to mitigate design basis accidents at CCNPP, the ILRT extension does not impact CDF. Therefore, the relevant risk-impact metric is LERF.

For CCNPP, 100% of the frequency of Class 3b sequences can be used as a very conservative first-order estimate to approximate the potential increase in LERF from the ILRT interval extension (consistent with the EPRI guidance methodology). Based on a 10-year test interval from Tables 5-16 and 5-17, the Class 3b frequency is $1.17E-7$ /year for Unit 1 and $1.02E-7$ for Unit 2; based on a 15-year test interval from Tables 5-18 and 5-19, the Class 3b frequency is $1.76E-7$ for Unit 1 and $1.54E-7$ for Unit 2. Thus, the increase in the overall probability of LERF due to Class 3b sequences that is due to increasing the ILRT test interval from 3 to 15 years is $1.41E-7$ /year for Unit 1 and $1.23E-7$ for Unit 2. Similarly, the increase due to increasing the interval from 10 to 15 years is $5.87E-8$ /year for Unit 1 and $5.12E-8$ for Unit 2. As can be seen, even with the conservatisms included in the evaluation (per the EPRI methodology), the estimated change in LERF is below the threshold criteria for a small change when comparing the 15-year results to the current 10-year requirement, and slightly greater than the criteria when compared to the original 3-year requirement. Table 5-20 summarizes these results.

Table 5-20 – Impact on LERF due to Extended Type A Testing Intervals

ILRT Inspection Interval	Unit 1: 3 Years (baseline)	Unit 1: 10 Years	Unit 1: 15 Years	Unit 2: 3 Years (baseline)	Unit 2: 10 Years	Unit 2: 15 Years
Class 3b (Type A LERF)	3.52E-08	1.17E-07	1.76E-07	3.07E-08	1.02E-07	1.54E-07
ΔLERF (3 year baseline)		8.22E-08	1.41E-07		7.16E-08	1.23E-07
ΔLERF (10 year baseline)			5.87E-08			5.12E-08

The increase in the overall probability of LERF due to Class 3b sequences being slightly greater than $1E-7$ is not unexpected. Since the target is exceeded, some refinement is necessary. One method to remove some conservatism is to examine the source term expected to be available for release during the accident sequence. The source term is greatly reduced if the debris expelled from the reactor remains covered with water. Therefore, if the accident sequence contains containment spray success, the source term is not considered to lead to a large early release. The methodology developed in Reference 33 is used for this containment spray success sensitivity. Excluding INTACT scenarios where containment spray is credited and therefore scrubbing the source term release results in a frequency reduction.

Conservatisms are further reduced by analyzing the source term release times. Early release timing is defined by time short enough that ability to evacuate nearby population is impaired such that a fatality is possible. Reference 36 contains the development of Evacuation Time Estimates for the Calvert Cliffs Nuclear Power Plant. Review of this report shows that the average evacuation time is estimated to be 6.5 hours. Therefore, for this assessment, an early

release is defined as occurring before 6.5 hours. By reviewing CCNPP's MAAP runs in the Level 2 severe accident report [Reference 18], it was determined three cases had source terms released after the 6.5 hour mark. The first case is HRIF, which simulates a loss of main feedwater due to a station blackout (SBO). The last two cases, GIOY and MRIF, evaluate small LOCAs inside containment. These three MAAP cases are matched with a corresponding plant damage state (PDS) in CCNPP's Level 2 notebook [Reference 32]. Table 5-21 displays CCNPP's PDSs.

PDS	Containment Bypass?	RCS Pressure at Time of Core Damage?	Feedwater Availability?	Pressurizer PORV/SRV Status?	CHR?	AC Power Available?
1	No	High	Not available	Not stuck open	Not Available	Available
4	No	Low	Available	Not stuck open	Not Available	Available
5	No	High	Available	Not stuck open	Not Available	Available
6	No	Low	Available	Not stuck open	Available	Available
7	SGTR	N/A	N/A	N/A	N/A	N/A
8	ISLOCA	N/A	N/A	N/A	N/A	N/A
9	No	High	Available	Not stuck open	Not Available	Not Available
10	No	High	Not Available	Not stuck open	Available	Available
14	No	High	Not Available	Stuck open	Not Available	Not Available
15	No	High	Not Available	Not stuck open	Not Available	Not Available
16	No	High	Available	Stuck open	Not Available	Not Available
17	No	Low	Available	Not Stuck Open	Not Available	Not Available

The HRIF MAAP case models a SBO that leads to a loss of main feedwater. The analysis assumes a loss of containment heat removal and AC power. The reactor coolant system is isolated and the containment remains intact. Core damage occurs while the reactor coolant system is at high pressure. Based on the information in Table 5-21, this case can be used to represent PDS 15. Table 2-2 of Reference 32 contains the list of all the Level 1 core damage

accident sequences and how each is mapped to a PDS. Using the correlation of the HRIF case and the SBO cases that contain a loss of feedwater (PDS 15), it is determined that the frequency contribution can be removed from the LERF contribution because the release occurs at approximately 36 hours, which is greater than 6.5 hours. The impacted sequences are SBO004, SBO005, SBO010, SBO013, SBO015, SBO018, SBO019, and SBO039.

Another MAAP case evaluated is GIOY, which involves a Small LOCA inside containment with an equivalent break size of 0.005 ft² and the containment isolated. The reactor coolant system is at high pressure with auxiliary feedwater (AFW) and AC power available; containment air cooling (CAC) provides containment cooling and maintains containment pressure. The radionuclide release occurs at approximately 8 hours [Reference 18], which is greater than 6.5 hours. Therefore, this case can be used to represent PDS 5. The following sequences are Small LOCA cases assigned to PDS 5 and are excluded based on their late release: SLOCA002, SLOCA003, and SLOCA012.

Another MAAP case evaluated is MRIF, which involves a Small LOCA inside containment with an equivalent break size of 0.02 ft². Containment is isolated; the reactor coolant pressure is high; AFW and containment heat removal are not available; AC power is available. These characteristics map to PDS 1. The release occurs at approximately 28 hours [Reference 18] which is greater than 6.5 hours. The following sequences are Small LOCA cases assigned to PDS 5 and are removed from the PDS 1 frequency: TRAN003, TRAN004, TRAN005, TRAN007, TRAN008, TRAN009, SLOCA007, and SLOCA011 [Reference 18].

In addition to the late release timing associated with each of these MAAP cases, the release fractions of the iodine group are less than 1E-03 in all cases. Typically, a release fraction of 10% of iodine is used to represent a large release and thus none of the releases associated with these cases can be considered large. For ease of review, the release timing and fractions from Reference 18 are contained in Attachment 2.

The exclusion of these frequencies yields new Level 2 results. Table 5-22 shows adjusted release category frequencies after some conservatisms from containment spray success and release timing are excluded.

Release Category	EPRI Category	Unit 1 Frequency (/yr)	Unit 2 Frequency (/yr)
INTACT	Class 1	3.28E-06	1.18E-06
LERF	Classes 2, 7, 8	9.51E-07	1.08E-06
SERF	Class 6	1.54E-06	5.74E-07
LATE	Class 1	2.37E-06	2.18E-06 ¹
Total (CDF)	N/A	8.14E-06	5.01E-06

1. Unit 2 LATE was quantified at 5E-12 truncation. The other end states were quantified at 1E-12.

Substituting these values into the previously defined equations and calculation method yields the final results displayed in Table 5-23.

Table 5-23 – Impact on LERF due to Extended Type A Testing Intervals with Adjusted CDF

ILRT Inspection Interval	Unit 1: 3 Years (baseline)	Unit 1: 10 Years	Unit 1: 15 Years	Unit 2: 3 Years (baseline)	Unit 2: 10 Years	Unit 2: 15 Years
Class 3b (Type A LERF)	1.14E-08	3.78E-08	5.68E-08	6.14E-09	2.05E-08	3.07E-08
ΔLERF (3 year baseline)		2.65E-08	4.54E-08		1.43E-08	2.46E-08
ΔLERF (10 year baseline)			1.89E-08			1.02E-08

The adjusted containment spray and PDS inputs allow the Unit 1 and 2 values to be much less than the 1E-7 LERF metric. The delta LERF between the 3 years and the 15 years is 4.54E-8/yr for Unit 1 and 2.46E-8/yr for Unit 2. These values show that the proposed extension meets the definition of a very small change in risk as defined in Regulatory Guide 1.174.

5.2.5 Step 5 – Determine the Impact on the Conditional Containment Failure Probability (CCFP)

Another parameter that the NRC guidance in RG 1.174 [Reference 4] states can provide input into the decision-making process is the change in the conditional containment failure probability (CCFP). The CCFP is defined as the probability of containment failure given the occurrence of an accident. This probability can be expressed using the following equation:

$$CCFP = 1 - \frac{f(ncf)}{CDF}$$

where $f(ncf)$ is the frequency of those sequences that do not result in containment failure; this frequency is determined by summing the Class 1 and Class 3a results.

Since CCFP is only concerned with a containment failure and not whether the release is small or large, the Class 1 results without refinement must be used to calculate the CCFP. Table 5-24 shows the steps and results of this calculation. The difference in CCFP between the 3-year test interval and 15-year test interval is 0.88% for Unit 1 and 0.87% for Unit 2.

Table 5-24 – Impact on CCFP due to Extended Type A Testing Intervals

ILRT Inspection Interval	Unit 1: 3 Years (baseline)	Unit 1: 10 Years	Unit 1: 15 Years	Unit 2: 3 Years (baseline)	Unit 2: 10 Years	Unit 2: 15 Years
$f(ncf)$ (/yr)	1.28E-05	1.27E-05	1.27E-05	1.13E-05	1.12E-05	1.12E-05
$f(ncf)/CDF$	0.796	0.791	0.787	0.799	0.794	0.791
CCFP	0.204	0.209	0.213	0.201	0.206	0.209
ΔCCFP (3 year baseline)		0.511%	0.876%		0.508%	0.871%
ΔCCFP (10 year baseline)			0.365%			0.363%

As stated in Section 2.0, a change in the CCFP of up to 1.5% is assumed to be small. The increase in the CCFP from the 3 in 10 year interval to 1 in 15 year interval is 0.876% for Unit 1 and 0.871% for Unit 2. Therefore, this increase is judged to be very small.

5.3 Sensitivities

5.3.1 Potential Impact from External Events Contribution

An assessment of the impact of external events is performed. The primary basis for this investigation is the determination of the total LERF following an increase in the ILRT testing interval from 3 in 10 years to 1 in 15 years.

Calvert Cliffs is transitioning to NFPA 805 licensing basis for fire protection and submitted a License Amendment Request (LAR) on September 24, 2013 (ADAMS Accession No. ML13301A673). This transition included performing a Fire PRA and committing to modifications to reduce the fire-induced core damage and large early release frequencies to those reported in the NFPA 805 LAR. Compensatory actions have been implemented to reduce the fire risk until the modifications are implemented. The Unit 1 ILRT is scheduled for 2016, which is prior to the scheduled implementation of all the modifications by 2018. It is anticipated that many, but not all, of the NFPA 805 modifications will be completed by the Unit 1 refueling outage. Risk mitigation strategies will be in place for any open modification. These strategies may be actions to reduce fire initiating event probabilities, actions to improve suppression probability, and/or actions to recover or protect systems that mitigate core damage and large early release accident sequences. The Unit 2 ILRT is scheduled for 2023, so the NFPA 805 modifications will be implemented prior to the extension. The section evaluates the fire risk using the Fire PRA. Section 5.3.1.1 uses the IPEEE fire risk values to evaluate fire risk.

The Fire PRA model 6.1M was used to obtain the fire CDF and LERF values. To reduce conservatism in the model, the plant damage state methodology described in Section 5.2.4 was also applied to the CDF portion of the Fire PRA model. The following shows the calculation for Class 3b for Units 1 and 2:

$$Freq_{U1class3b} = P_{class3b} * (CDF1 - PDS_{CDF1}) = \frac{0.5}{218} * (3.18E-05 - 2.20E-5) = 2.25E-8$$

$$Freq_{U2class3b} = P_{class3b} * (CDF2 - PDS_{CDF2}) = \frac{0.5}{218} * (3.62E-05 - 2.26E-5) = 3.12E-8$$

$$Freq_{U1class3b10yr} = \frac{10}{3} * P_{class3b} * (CDF1 - PDS_{CDF1}) = \frac{10}{3} * \frac{0.5}{218} * (3.18E-05 - 2.20E-5) = 7.49E-8$$

$$Freq_{U2class3b10yr} = \frac{10}{3} * P_{class3b} * (CDF2 - PDS_{CDF2}) = \frac{10}{3} * \frac{0.5}{218} * (3.62E-05 - 2.26E-5) = 1.04E-07$$

$$Freq_{U1class3b15yr} = \frac{15}{3} * P_{class3b} * (CDF1 - PDS_{CDF1}) = 5 * \frac{0.5}{218} * (3.18E-05 - 2.20E-5) = 1.12E-07$$

$$Freq_{U2class3b15yr} = \frac{15}{3} * P_{class3b} * (CDF2 - PDS_{CDF2}) = 5 * \frac{0.5}{218} * (3.62E-05 - 2.26E-5) = 1.56E-07$$

Seismic events were addressed through a simplified seismic PRA in Section 3 of the IPEEE for CCNPP [Reference 35]. The Seismic PRA method screened all the components that met a high confidence low probability of failure (HCLPF) for the review level seismic event occurring with a magnitude of 0.3g. The remaining components were grouped together as a proxy component. It was assumed that if this proxy component failed it would result in core damage. This method is considered conservative.

Data from Table 3-6 of the IPEEE Seismic Analysis [Reference 35] is used to calculate a Class 3b frequency due to seismic. As noted in Table 3-6 of the IPEEE Seismic Analysis, the values given in Table 3-6 reflect quantification without the surrogate top event LA. Top event LA represents seismic failure of rugged plant systems at a conservative screening fragility. Therefore, the total CDF is higher than the 1.07E-05/yr value given in Table 3-6. The CDF

values given in Section 8.1 of the IPEEE Seismic Analysis [Reference 35] are 1.29E-5/yr for Unit 1 and 1.52E-5/yr for Unit 2. The CDF contribution from surrogate top event LA was not included in the Unit 1 containment failure frequencies provided in the IPEEE (no containment failure frequencies are provided for Unit 2). To conservatively add in the difference between the CDF including surrogate top event LA and the total Unit 1 CDF of 1.07E-5/yr from Table 3-6 of the IPEEE for the Class 3b frequency calculation, it is added to Containment Category I Failure (Intact). Note that the Intact category CDF is slightly rounded so that the total seismic CDF is preserved. Then, the percent each category contributes to the total CDF is calculated for the Unit 1 values and applied to the Unit 2 values because it is assumed that Unit 2 would have similar containment failure fractions to Unit 1.

Table 5-25 – Seismic Contribution to Frequencies of Containment Failure Categories

Containment Failure Category	Unit 1 Seismic CDF (/yr)	Percent of CDF	Unit 2 Seismic CDF (/yr)
I. Intact Containment	2.69E-06	20.85%	3.17E-06
II. Late Containment Failure	8.63E-06	66.90%	1.02E-05
III. Early Small Containment Failure	1.70E-07 to 1.27E-06	1.32% to 9.84%	2.00E-07 to 1.50E-06
IV. Early Large Containment Failure	3.13E-07 to 1.41E-06	2.43% to 10.93%	3.69E-07 to 1.66E-06
V. Small Containment Bypass	0	0%	0
VI. Large Containment Bypass	0	0%	0
Total	1.29E-05		1.52E-05

Note: The Seismic contribution to Containment failure categories III and IV is shown as a range of values. A range is shown because the contribution of a certain PDS will be apportioned between the small and large early containment failures, but the ratio is unknown. Therefore, we show a range of values which reflect the contribution of this PDS from being attributed entirely to early-large containment failures (conservative) to early-small containment failures. See section 3.1.6.1 of the IPEEE Seismic Analysis for a more detailed explanation.

Using this seismic data, the Class 3b frequency can be calculated by the following formulas:

$$Freq_{U1class3b} = P_{class3b} * (CDF - CatIV - CatVI) = \frac{0.5}{218} * (1.29E-5 - 3.13E-7 - 0) = 2.89E-8$$

$$Freq_{U2class3b} = P_{class3b} * (CDF - CatIV - CatVI) = \frac{0.5}{218} * (1.52E-5 - 3.69E-7 - 0) = 3.40E-8$$

$$Freq_{U1class3b10yr} = \frac{10}{3} * P_{class3b} * (CDF - CatIV - CatVI) = \frac{10}{3} * \frac{0.5}{218} * (1.29E-5 - 3.13E-7 - 0) = 9.62E-8$$

$$Freq_{U2class3b10yr} = \frac{10}{3} * P_{class3b} * (CDF - CatIV - CatVI) = \frac{10}{3} * \frac{0.5}{218} * (1.52E-5 - 3.69E-7 - 0) = 1.13E-7$$

$$Freq_{U1class3b15yr} = \frac{15}{3} * P_{class3b} * (CDF - CatIV - CatVI) = \frac{15}{3} * \frac{0.5}{218} * (1.29E-5 - 3.13E-7 - 0) = 1.44E-7$$

$$Freq_{U2class3b15yr} = \frac{15}{3} * P_{class3b} * (CDF - CatIV - CatVI) = \frac{15}{3} * \frac{0.5}{218} * (1.52E-5 - 3.69E-7 - 0) = 1.70E-7$$

CNNPP topographical location presents the opportunity for high wind events. These events include tornadoes, thunderstorms, freezing precipitation, and hurricanes. Hurricanes pose approximately one threat per year and one significant threat per 10 years (Reference 24). These natural disasters are modeled in the internal events model. As shown in Table 5-2 and 5-4 show that high wind risk is approximately two orders of magnitude lower than fire risk. Since high wind

risk is already included in the internal events PRA, no further analysis is necessary to include its contribution to Class 3b frequency.

The seismic and fire contributions to Class 3b frequencies are then combined to obtain the total external event contribution to Class 3b frequencies. The change in LERF is calculated for the 1 in 10 year and 1 in 15 year cases and the change defined for the external events in Tables 5-26 and 5-27 for Units 1 and 2, respectively.

Table 5-26 – CCNPP Unit 1 External Event Impact on ILRT LERF Calculation

Hazard	EPRI Accident Class 3b Frequency			LERF Increase (from 3 per 10 years to 1 per 15 years)
	3 per 10 year	1 per 10 year	1 per 15 years	
External Events	5.13E-08	1.71E-07	2.57E-07	2.05E-07
Internal Events	1.14E-08	3.78E-08	5.68E-08	4.54E-08
Combined	6.27E-08	2.09E-07	3.13E-07	2.51E-07

Table 5-27 – CCNPP Unit 2 External Event Impact on ILRT LERF Calculation

Hazard	EPRI Accident Class 3b Frequency			LERF Increase (from 3 per 10 years to 1 per 15 years)
	3 per 10 year	1 per 10 year	1 per 15 years	
External Events	6.52E-08	2.17E-07	3.26E-07	2.61E-07
Internal Events	6.14E-09	2.05E-08	3.07E-08	2.46E-08
Combined	7.13E-08	2.38E-07	3.57E-07	2.85E-07

The internal event results are also provided to allow a composite value to be defined. When both the internal and external event contributions are combined the total change in Unit 1 and 2 LERF meet the guidance for small change in risk, as it exceeds the 1.0E-7/yr and remains less than 1.0E-6 change in LERF for both units. For this change in LERF to be acceptable, total LERF must be less than 1.0E-5.

Conservatively using the highest seismic LERF value and not crediting containment spray success or plant damage state adjustments for the Internal Events or Fire PRA, the total LERF values are calculated below:

$$\begin{aligned} \text{Unit 1: LERF}_{U1} &= \text{LERF}_{U1\text{internal}} + \text{LERF}_{U1\text{seismic}} + \text{LERF}_{U1\text{fire}} + \text{LERF}_{U1\text{class3Bincrease}} \\ &= 1.39\text{E-}6/\text{yr} + 1.41\text{E-}6/\text{yr} + 2.97\text{E-}6/\text{yr} + 2.51\text{E-}07/\text{yr} = 6.02\text{E-}6/\text{yr} \end{aligned}$$

$$\begin{aligned} \text{Unit 2: LERF}_{U2} &= \text{LERF}_{U2\text{internal}} + \text{LERF}_{U2\text{seismic}} + \text{LERF}_{U2\text{fire}} + \text{LERF}_{U2\text{class3Bincrease}} \\ &= 1.56\text{E-}6/\text{yr} + 1.66\text{E-}6/\text{yr} + 4.17\text{E-}6/\text{yr} + 2.85\text{E-}07/\text{yr} = 7.68\text{E-}6/\text{yr} \end{aligned}$$

Since the total LERF for both units is less than 1.0E-5, it is acceptable for the ΔLERF to be between 1.0E-7 and 1.0E-6.

5.3.1.1 Potential Impact from External Events Contribution Using IPEEE Fire Analysis

An assessment of the impact of external events is also performed using fire risk analysis from the IPEEE [Reference 35] rather than the Fire PRA model 6.1M. Table 4.7 from the simplified IPEEE fire PRA shows the frequencies of major containment failure categories for Unit 1

[Reference 35]. The same containment failure category percentages are assumed for Unit 2; as given in Section 8.2 of the IPEEE, the estimated Unit 2 fire CDF is 9.6E-5/yr. The Level 2 results are shown here in table 5-28.

Containment Failure Category	Percentage	Unit 1 Fire CDF (/yr)	Unit 2 Fire CDF (/yr)
I. Intact Containment	36.4%	2.67E-05	3.50E-05
II. Late Containment Failure	55.5%	4.07E-05	5.33E-05
III. Early Small Containment Failure	1.7%	1.21E-06	1.58E-06
IV. Early Large Containment Failure	6.5%	4.67E-06	6.13E-06
V. Small Containment Bypass	0.0%	0.00E+00	0.00E+00
VI. Large Containment Bypass	0.0%	0.00E+00	0.00E+00
Total		7.32E-05	9.60E-05

Note: the Unit 1 CDF obtained by summing the CDF values given in Table 4.7 of the IPEEE [Reference 35] is 7.32E-05, as cited in this table. This number differs from 7.29E-05 cited in Section 8 of the IPEEE [Reference 35]. The CDF value used in this table and subsequent calculations is conservatively chosen as 7.32E-05. Also, the percentages given in Table 4.7 of the IPEEE [Reference 35] sum to 101.8%. These percentages are adjusted based on the individual values of the Unit 1 Containment Failure Categories to sum to 100.0%.

Using the IPEEE fire data, the Class 3b frequency can be calculated by the following formulas:

$$\text{Unit 1: } \text{Freq}_{U1class3b} = P_{class3b} * (CDF - \text{CatIV} - \text{CatVI}) = \frac{0.5}{218} * (7.32E-5 - 4.67E-6 - 0) = 1.57E-7$$

$$\begin{aligned} \text{Unit 1: } \text{Freq}_{U1class3b10yr} &= \frac{10}{3} * P_{class3b} * (CDF - \text{CatIV} - \text{CatVI}) \\ &= \frac{10}{3} * \frac{0.5}{218} * (7.32E-5 - 4.67E-6 - 0) = 5.24E-7 \end{aligned}$$

$$\begin{aligned} \text{Unit 1: } \text{Freq}_{U1class3b15yr} &= \frac{15}{3} * P_{class3b} * (CDF - \text{CatIV} - \text{CatVI}) \\ &= 5 * \frac{0.5}{218} * (7.32E-5 - 4.67E-6 - 0) = 7.86E-7 \end{aligned}$$

$$\text{Unit 2: } \text{Freq}_{U2class3b} = P_{class3b} * (CDF - \text{CatIV} - \text{CatVI}) = \frac{0.5}{218} * (9.60E-5 - 6.13E-6 - 0) = 2.06E-7$$

$$\begin{aligned} \text{Unit 2: } \text{Freq}_{U2class3b10yr} &= \frac{10}{3} * P_{class3b} * (CDF - \text{CatIV} - \text{CatVI}) \\ &= \frac{10}{3} * \frac{0.5}{218} * (9.60E-5 - 6.13E-6 - 0) = 6.87E-7 \end{aligned}$$

$$\begin{aligned} \text{Unit 2: } \text{Freq}_{U2class3b15yr} &= \frac{15}{3} * P_{class3b} * (CDF - \text{CatIV} - \text{CatVI}) \\ &= 5 * \frac{0.5}{218} * (9.60E-5 - 6.13E-6 - 0) = 1.03E-6 \end{aligned}$$

As done in Section 5.3.1, the IPEEE seismic and fire contributions to Class 3b frequencies are then combined to obtain the total external event contribution to Class 3b frequencies. The change in LERF is calculated for the 1 in 10 year and 1 in 15 year cases and the change defined for the external events in Tables 5-29 and 5-30 for Units 1 and 2, respectively.

Table 5-29 – CCNPP Unit 1 External Event Impact on ILRT LERF Calculation

Hazard	EPRI Accident Class 3b Frequency			LERF Increase (from 3 per 10 years to 1 per 15 years)
	3 per 10 year	1 per 10 year	1 per 15 years	
External Events	1.86E-07	6.20E-07	9.30E-07	7.44E-07
Internal Events	1.14E-08	3.78E-08	5.68E-08	4.54E-08
Combined	1.97E-07	6.58E-07	9.87E-07	7.89E-07

Table 5-30 – CCNPP Unit 2 External Event Impact on ILRT LERF Calculation

Hazard	EPRI Accident Class 3b Frequency			LERF Increase (from 3 per 10 years to 1 per 15 years)
	3 per 10 year	1 per 10 year	1 per 15 years	
External Events	2.40E-07	8.00E-07	1.20E-06	9.61E-07
Internal Events	6.14E-09	2.05E-08	3.07E-08	2.46E-08
Combined	2.46E-07	8.21E-07	1.23E-06	9.88E-07

The internal event results are also provided to allow a composite value to be defined. When both the internal and external event contributions are combined the total change in Unit 1 and 2 LERF meet the guidance for small change in risk, as it exceeds the 1.0E-7/yr and remains less than 1.0E-6 change in LERF for both units. For this change in LERF to be acceptable, total LERF must be less than 1.0E-5.

Conservatively using the highest seismic LERF value and not crediting containment spray success or plant damage state adjustments for the Internal Events PRA, the total LERF values are calculated below:

$$\begin{aligned} \text{Unit 1: } LERF_{U1} &= LERF_{U1\text{internal}} + LERF_{U1\text{seismic}} + LERF_{U1\text{fire}} + LERF_{U1\text{class3Bincrease}} \\ &= 1.39E-6/\text{yr} + 1.41E-6/\text{yr} + 4.67E-6/\text{yr} + 7.89E-7/\text{yr} = 8.26E-6/\text{yr} \end{aligned}$$

$$\begin{aligned} \text{Unit 2: } LERF_{U2} &= LERF_{U2\text{internal}} + LERF_{U2\text{seismic}} + LERF_{U2\text{fire}} + LERF_{U2\text{class3Bincrease}} \\ &= 1.56E-6/\text{yr} + 1.66E-6/\text{yr} + 6.13E-6/\text{yr} + 9.84E-7/\text{yr} = 1.03E-5/\text{yr} \end{aligned}$$

The Unit 2 LERF is barely greater than 1.0E-5. However, the Unit 2 Seismic LERF is between 3.69E-07 and 1.66E-06, and the highest Seismic LERF value was conservatively used to calculate 1.03E-5. If the 74th percentile or smaller value of this range ($\leq 1.32E-6$) is used, the total Unit 2 LERF is less than 1.0E-5. Moreover, the IPEEE does not include recent significant plant modifications designed specifically to reduce fire risk. Therefore, it is reasonable to conclude that the total Unit 2 LERF is less than 1.0E-5. Since the total LERF for both units is less than 1.0E-5, it is acceptable for the Δ LERF to be between 1.0E-7 and 1.0E-6.

5.3.2 Potential Impact from Steel Liner Corrosion Likelihood

A quantitative assessment of the contribution of steel liner corrosion likelihood impact was performed for the risk impact assessment for extended ILRT intervals. As a sensitivity run, the internal event CDF was used to calculate the Class 3b frequency. The impact on the Class 3b frequency due to increases in the ILRT surveillance interval was calculated for steel liner

corrosion likelihood using the relationships described in Section 5.1.4. The EPRI Category 3b frequencies for the 3 per 10-year, 10-year and 15-year ILRT intervals were quantified using the internal events CDF. The change in the LERF, change in CCFP, and change in Annual Dose Rate due to extending the ILRT interval from 3 in 10 years to 1 in 10 years, or to 1 in 15 years are provided in Tables 5-31 – 5-36. Since CCFP is only concerned with a containment failure and not whether the release is small or large, the Class 1 results without containment spray and PDS refinement is used to calculate the CCFP. The Annual Dose Rate calculations are performed using the containment spray and PDS adjustments. The steel liner corrosion likelihood was increased by a factor of 1000, 10000, and 100000. Except for extreme factors of 10000 and 100000, the corrosion likelihood is relatively insensitive to the results.

Table 5-31 – Unit 1 Steel Liner Corrosion Sensitivity Cases

	3b Frequency (3-per-10 year ILRT)	3b Frequency (1-per-10 year ILRT)	3b Frequency (1-per-15 year ILRT)	LERF Increase (3-per-10 to 1-per-10)	LERF Increase (3-per-10 to 1-per-15)	LERF Increase (1-per-10 to 1-per-15)
Internal Event 3B Contribution	1.14E-08	3.78E-08	5.68E-08	2.65E-08	4.54E-08	1.89E-08
Corrosion Likelihood X 1000	1.15E-08	3.98E-08	6.36E-08	2.84E-08	5.22E-08	2.38E-08
Corrosion Likelihood X 10000	1.24E-08	5.76E-08	1.25E-07	4.52E-08	1.13E-07	6.77E-08
Corrosion Likelihood X 100000	2.17E-08	2.35E-07	7.42E-07	2.14E-07	7.20E-07	5.06E-07

Table 5-32 – Unit 2 Steel Liner Corrosion Sensitivity Cases

	3b Frequency (3-per-10 year ILRT)	3b Frequency (1-per-10 year ILRT)	3b Frequency (1-per-15 year ILRT)	LERF Increase (3-per-10 to 1-per-10)	LERF Increase (3-per-10 to 1-per-15)	LERF Increase (1-per-10 to 1-per-15)
Internal Event 3B Contribution	6.14E-09	2.05E-08	3.07E-08	1.43E-08	2.46E-08	1.02E-08
Corrosion Likelihood X 1000	6.19E-09	2.15E-08	3.44E-08	1.53E-08	2.82E-08	1.29E-08
Corrosion Likelihood X 10000	6.70E-09	3.11E-08	6.77E-08	2.44E-08	6.10E-08	3.66E-08
Corrosion Likelihood X 100000	1.17E-08	1.27E-07	4.01E-07	1.16E-07	3.89E-07	2.74E-07

Table 5-33 – Unit 1 Steel Liner Corrosion Sensitivity Cases

	CCFP (3-per-10 year ILRT)	CCFP (1-per-10 year ILRT)	CCFP (1-per-15 year ILRT)	CCFP Increase (3-per-10 to 1-per-10)	CCFP Increase (3-per-10 to 1-per-15)	CCFP Increase (1-per-10 to 1-per-15)
Baseline CCFP	2.04E-01	2.09E-01	2.13E-01	5.11E-03	8.76E-03	3.65E-03
Corrosion Likelihood X 1000	2.04E-01	2.09E-01	2.13E-01	5.16E-03	8.84E-03	3.68E-03
Corrosion Likelihood X 10000	2.04E-01	2.10E-01	2.14E-01	5.57E-03	9.56E-03	3.98E-03
Corrosion Likelihood X 100000	2.06E-01	2.16E-01	2.23E-01	9.76E-03	1.67E-02	1.29E-02

Table 5-34 – Unit 2 Steel Liner Corrosion Sensitivity Cases

	CCFP (3-per-10 year ILRT)	CCFP (1-per-10 year ILRT)	CCFP (1-per-15 year ILRT)	CCFP Increase (3-per-10 to 1-per-10)	CCFP Increase (3-per-10 to 1-per-15)	CCFP Increase (1-per-10 to 1-per-15)
Baseline CCFP	2.01E-01	2.06E-01	2.09E-01	5.08E-03	8.71E-03	3.63E-03
Corrosion Likelihood X 1000	2.01E-01	2.06E-01	2.09E-01	5.13E-03	8.79E-03	3.66E-03
Corrosion Likelihood X 10000	2.01E-01	2.06E-01	2.10E-01	5.54E-03	9.50E-03	3.96E-03
Corrosion Likelihood X 100000	2.03E-01	2.12E-01	2.19E-01	9.70E-03	1.66E-02	6.93E-03

Table 5-35 – Unit 1 Steel Liner Corrosion Sensitivity Cases

	Dose Rate (3-per-10 year ILRT)	Dose Rate (1-per-10 year ILRT)	Dose Rate (1-per-15 year ILRT)	Dose Rate Increase (3-per-10 to 1-per-10)	Dose Rate Increase (3-per-10 to 1-per-15)	Dose Rate Increase (1-per-10 to 1-per-15)
Dose Rate	3.63E-02	1.21E-01	1.82E-01	8.48E-02	1.45E-01	6.06E-02
Corrosion Likelihood X 1000	3.67E-02	1.27E-01	2.04E-01	9.08E-02	1.67E-01	7.61E-02
Corrosion Likelihood X 10000	3.96E-02	1.84E-01	4.01E-01	1.45E-01	3.61E-01	2.17E-01
Corrosion Likelihood X 100000	6.94E-02	7.53E-01	2.37E+00	6.84E-01	2.30E+00	1.62E+00

Table 5-36 – Unit 2 Steel Liner Corrosion Sensitivity Cases

	CCFP (3-per-10 year ILRT)	CCFP (1-per-10 year ILRT)	CCFP (1-per-15 year ILRT)	CCFP Increase (3-per-10 to 1-per-10)	CCFP Increase (3-per-10 to 1-per-15)	CCFP Increase (1-per-10 to 1-per-15)
Dose Rate	1.96E-02	6.55E-02	9.82E-02	4.58E-02	7.86E-02	3.27E-02
Corrosion Likelihood X 1000	1.98E-02	6.89E-02	1.10E-01	4.91E-02	9.02E-02	4.12E-02
Corrosion Likelihood X 10000	2.14E-02	9.96E-02	2.17E-01	7.82E-02	1.95E-01	1.17E-01
Corrosion Likelihood X 100000	3.75E-02	4.07E-01	1.28E+00	3.70E-01	1.25E+00	8.76E-01

5.3.3 Expert Elicitation Sensitivity

Another sensitivity case on the impacts of assumptions regarding pre-existing containment defect or flaw probabilities of occurrence and magnitude, or size of the flaw, is performed as described in Reference 24. In this sensitivity case, an expert elicitation was conducted to develop probabilities for pre-existing containment defects that would be detected by the ILRT only based on the historical testing data.

Using the expert knowledge, this information was extrapolated into a probability versus magnitude relationship for pre-existing containment defects. The failure mechanism analysis also used the historical ILRT data augmented with expert judgment to develop the results. Details of the expert elicitation process and results are contained in Reference 24. The expert elicitation process has the advantage of considering the available data for small leakage events, which have occurred in the data, and extrapolate those events and probabilities of occurrence to the potential for large magnitude leakage events.

The expert elicitation results are used to develop sensitivity cases for the risk impact assessment. Employing the results requires the application of the ILRT interval methodology using the expert elicitation to change the probability of pre-existing leakage in the containment.

The baseline assessment uses the Jefferys non-informative prior and the expert elicitation sensitivity study uses the results of the expert elicitation. In addition, given the relationship between leakage magnitude and probability, larger leakage that is more representative of large early release frequency, can be reflected. For the purposes of this sensitivity, the same leakage magnitudes that are used in the basic methodology (i.e., 10 La for small and 100 La for large) are used here. Table 5-37 presents the magnitudes and probabilities associated with the Jefferys non-informative prior and the expert elicitation use in the base methodology and this sensitivity case.

Table 5-37 – CCNPP Summary of ILRT Extension Using Expert Elicitation Values (from Reference 24)

Leakage Size (L _a)	Jefferys Non-Informative Prior	Expert Elicitation Mean Probability of Occurrence	Percent Reduction
10	2.70E-02	3.88E-03	86%
100	2.70E-03	9.86E-04	64%

Taking the baseline analysis and using the values provided in Tables 5-16 – 5-19 for the expert elicitation yields the results in Tables 5-38 and 5-39 for Units 1 and 2, respectively, are developed.

Table 5-38 – CCNPP Unit 1 Summary of ILRT Extension Using Expert Elicitation Values

Accident Class	ILRT Interval							
	3 per 10 Years				1 per 10 Years		1 per 15 Years	
	Base Frequency	Adjusted Base Frequency	Dose (person-rem)	Dose Rate (person-rem/yr)	Frequency	Dose Rate (person-rem/yr)	Frequency	Dose Rate (person-rem/yr)
1	1.28E-05	1.28E-05	3.40E+02	2.70E-04	1.26E-05	2.50E-04	1.25E-05	2.36E-04
2	5.01E-08	5.01E-08	2.00E+07	1.00E+00	5.01E-08	1.00E+00	5.01E-08	1.00E+00
3a	N/A	5.98E-08	3.40E+03	2.03E-04	1.99E-07	6.78E-04	2.99E-07	1.02E-03
3b	N/A	1.52E-08	3.40E+04	5.17E-04	5.06E-08	1.72E-03	7.60E-08	2.58E-03
6	1.87E-06	1.87E-06	7.01E+06	1.31E+01	1.87E-06	1.31E+01	1.87E-06	1.31E+01

Table 5-38 – CCNPP Unit 1 Summary of ILRT Extension Using Expert Elicitation Values

Accident Class	ILRT Interval							
	3 per 10 Years				1 per 10 Years		1 per 15 Years	
	Base Frequency	Adjusted Base Frequency	Dose (person-rem)	Dose Rate (person-rem/yr)	Frequency	Dose Rate (person-rem/yr)	Frequency	Dose Rate (person-rem/yr)
7	6.49E-07	6.49E-07	5.61E+07	3.64E+01	6.49E-07	3.64E+01	6.49E-07	3.64E+01
8	6.77E-07	6.77E-07	2.25E+07	1.52E+01	6.77E-07	1.52E+01	6.77E-07	1.52E+01
Totals	1.61E-05	1.61E-05	1.06E+08	6.58E+01	1.61E-05	6.58E+01	1.61E-05	6.58E+01
ΔLERF (3 per 10 yrs base)	N/A				3.55E-08		6.08E-08	
ΔLERF (1 per 10 yrs base)	N/A				N/A		2.53E-08	
CCFP	20.26%				20.48%		20.64%	

Table 5-39 – CCNPP Unit 2 Summary of ILRT Extension Using Expert Elicitation Values

Accident Class	ILRT Interval							
	3 per 10 Years				1 per 10 Years		1 per 15 Years	
	Base Frequency	Adjusted Base Frequency	Dose (person-rem)	Dose Rate (person-rem/yr)	Frequency	Dose Rate (person-rem/yr)	Frequency	Dose Rate (person-rem/yr)
1	1.13E-05	1.12E-05	3.40E+02	3.81E-03	1.10E-05	3.73E-03	1.08E-05	3.67E-03
2	4.34E-08	4.34E-08	2.00E+07	8.67E-01	4.34E-08	8.67E-01	4.34E-08	8.67E-01
3a	N/A	5.21E-08	3.40E+03	1.77E-04	1.74E-07	5.90E-04	2.60E-07	8.86E-04
3b	N/A	5.21E-08	3.40E+04	1.77E-03	1.74E-07	5.90E-03	2.60E-07	8.86E-03
6	1.25E-06	1.25E-06	7.01E+06	8.75E+00	1.25E-06	8.75E+00	1.25E-06	8.75E+00
7	8.35E-07	8.35E-07	5.61E+07	4.69E+01	8.35E-07	4.69E+01	8.35E-07	4.69E+01
8	6.72E-07	6.72E-07	2.25E+07	1.51E+01	6.72E-07	1.51E+01	6.72E-07	1.51E+01
Totals	1.41E-05	1.41E-05	1.06E+08	7.16E+01	1.41E-05	7.16E+01	1.41E-05	7.16E+01
ΔLERF (3 per 10 yrs base)	N/A				1.22E-07		2.09E-07	
ΔLERF (1 per 10 yrs base)	N/A				N/A		8.69E-08	
CCFP	20.21%				21.08%		21.69%	

The results illustrate how the expert elicitation reduces the overall change in LERF and the overall results are more favorable with regard to the change in risk.

5.3.4 Large Leak Probability Sensitivity Study

The large leak probability is a vital portion of determining the Class 3b frequency. CCNPP had previously calculated the large leak probability using the WCAP method. Table 5-40 presents the large leak probabilities for the baseline test, 10 year test interval, and 15 year test interval. Table 5-41 was developed using the same process as to calculate Class 3b.

Test Interval	WCAP Large Leak Probability	EPRI Accident Class 3b Frequency: Unit 1	EPRI Accident Class 3b Frequency: Unit 2
3 per 10 years	2.47E-4	1.38E-09	8.21E-10
10 years	7.41E-4	4.05E-09	2.41E-09
15 years	1.11E-3	5.96E-09	3.55E-09

Using the same EPRI approach, but with an updated Class 3b frequency calculated from the WCAP large leak probability data, Table 5-41 contains the final results for both units.

ILRT Inspection Interval	Unit 1: 3 Years (baseline)	Unit 1: 10 Years	Unit 1: 15 Years	Unit 2: 3 Years (baseline)	Unit 2: 10 Years	Unit 2: 15 Years
Class 3b (Type A LERF)	1.38E-09	4.05E-09	5.96E-09	8.21E-10	2.41E-09	3.55E-09
Δ LERF (3 year baseline)		2.67E-09	4.57E-09		1.59E-09	2.73E-09
Δ LERF (10 year baseline)			1.91E-09			1.14E-09

These results demonstrate that the EPRI methodology is conservative when used to calculate a large leak probability as compared to the WCAP method.

6.0 RESULTS

The results from this ILRT extension risk assessment for CCNPP are summarized in Table 6-1 for Unit 1 and Table 6-2 for Unit 2.

Table 6-1 – Unit 1 ILRT Extension Summary

Class	Dose (person-rem)	Base Case 3 in 10 Years		Extend to 1 in 10 Years		Extend to 1 in 15 Years	
		CDF/Year	Person-Rem/Year	CDF/Year	Person-Rem/Year	CDF/Year	Person-Rem/Year
1	3.20E+04	5.59E-06	1.79E-01	5.46E-06	1.75E-01	5.37E-06	1.72E-01
2	2.00E+07	3.33E-08	6.66E-01	3.33E-08	6.66E-01	3.33E-08	6.66E-01
3a	3.20E+05	4.56E-08	1.46E-02	1.52E-07	4.87E-02	2.28E-07	7.30E-02
3b	3.20E+06	1.14E-08	3.63E-02	3.78E-08	1.21E-01	5.68E-08	1.82E-01
6	7.01E+06	1.54E-06	1.08E+01	1.54E-06	1.08E+01	1.54E-06	1.08E+01
7	5.61E+07	2.49E-07	1.40E+01	2.49E-07	1.40E+01	2.49E-07	1.40E+01
8	2.25E+07	6.68E-07	1.50E+01	6.68E-07	1.50E+01	6.68E-07	1.50E+01
Total		8.14E-06	4.07E+01	8.14E-06	4.08E+01	8.14E-06	4.09E+01

ILRT Dose Rate from 3a and 3b

Δ Total Dose Rate	From 3 Years	N/A	1.15E-01	1.96E-01
	From 10 Years	N/A	N/A	8.18E-02
% Δ Dose Rate	From 3 Years	N/A	0.282%	0.483%
	From 10 Years	N/A	N/A	0.201%

3b Frequency (LERF)

Δ LERF	From 3 Years	N/A	2.65E-08	4.54E-08
	From 10 Years	N/A	N/A	1.89E-08

CCFP %

Δ CCFP%	From 3 Years	N/A	0.326%	0.558%
	From 10 Years	N/A	N/A	0.233%

Table 6-2 – Unit 2 ILRT Extension Summary

Class	Dose (person-rem)	Base Case 3 in 10 Years		Extend to 1 in 10 Years		Extend to 1 in 15 Years	
		CDF/Year	Person-Rem/Year	CDF/Year	Person-Rem/Year	CDF/Year	Person-Rem/Year
1	3.20E+04	3.32E-06	1.06E-01	3.25E-06	1.04E-01	3.20E-06	1.02E-01
2	2.00E+07	1.84E-08	3.69E-01	1.84E-08	3.69E-01	1.84E-08	3.69E-01
3a	3.20E+05	2.47E-08	7.89E-03	8.22E-08	2.63E-02	1.23E-07	3.95E-02
3b	3.20E+06	6.14E-09	1.96E-02	2.05E-08	6.55E-02	3.07E-08	9.82E-02
6	7.01E+06	5.74E-07	4.02E+00	5.74E-07	4.02E+00	5.74E-07	4.02E+00
7	5.61E+07	4.01E-07	2.25E+01	4.01E-07	2.25E+01	4.01E-07	2.25E+01
8	2.25E+07	6.60E-07	1.49E+01	6.60E-07	1.49E+01	6.60E-07	1.49E+01
Total		5.01E-06	4.19E+01	5.01E-06	4.19E+01	5.01E-06	4.20E+01

ILRT Dose Rate from 3a and 3b

Δ Total Dose Rate	From 3 Years	N/A	6.19E-02	1.06E-01
	From 10 Years	N/A	N/A	4.42E-02
% Δ Dose Rate	From 3 Years	N/A	0.148%	0.254%
	From 10 Years	N/A	N/A	0.106%

3b Frequency (LERF)

Δ LERF	From 3 Years	N/A	1.43E-08	2.46E-08
	From 10 Years	N/A	N/A	1.02E-08

CCFP %

Δ CCFP%	From 3 Years	N/A	0.286%	0.490%
	From 10 Years	N/A	N/A	0.204%

7.0 CONCLUSIONS AND RECOMMENDATIONS

Based on the results from Section 5.2 and the sensitivity calculations presented in Section 5.3, the following conclusions regarding the assessment of the plant risk are associated with extending the Type A ILRT test frequency to 15 years:

- Regulatory Guide 1.174 [Reference 4] provides guidance for determining the risk impact of plant-specific changes to the licensing basis. Regulatory Guide 1.174 defines very small changes in risk as resulting in increases of CDF less than $1.0E-06$ /year and increases in LERF less than $1.0E-07$ /year. Since the ILRT does not impact CDF, the relevant criterion is LERF. The increase in LERF resulting from a change in the Type A ILRT test interval from 3 in 10 years to 1 in 15 years is estimated as $4.54E-8$ /year for Unit 1 and $2.46E-8$ /year for Unit 2 using the EPRI guidance. As such, the estimated change in LERF is determined to be "very small" for both units using the acceptance guidelines of Regulatory Guide 1.174 [Reference 4].
- The effect resulting from changing the Type A test frequency to 1-per-15 years, measured as an increase to the total integrated plant risk for those accident sequences influenced by Type A testing, is 0.20 person-rem/year for Unit 1 and 0.11 person-rem/year for Unit 2. EPRI Report No. 1009325, Revision 2-A [Reference 24] states that a very small population dose is defined as an increase of ≤ 1.0 person-rem per year, or $\leq 1\%$ of the total population dose, whichever is less restrictive for the risk impact assessment of the extended ILRT intervals. This results of this calculation meet these criteria for both units. Moreover, the risk impact for the ILRT extension when compared to other severe accident risks is negligible.
- The increase in the conditional containment failure from the 3 in 10 year interval to 1 in 15 year interval is 0.558% for Unit 1 and 0.490% for Unit 2. EPRI Report No. 1009325, Revision 2-A [Reference 24] states that increases in CCFP of $\leq 1.5\%$ is very small. Therefore, this increase is judged to be very small.

Therefore, increasing the ILRT interval to 15 years is considered to be insignificant since it represents a very small change to the CCNPP risk profile.

Previous Assessments

The NRC in NUREG-1493 [Reference 6] has previously concluded that:

- Reducing the frequency of Type A tests (ILRTs) from 3 per 10 years to 1 per 20 years was found to lead to an imperceptible increase in risk. The estimated increase in risk is very small because ILRTs identify only a few potential containment leakage paths that cannot be identified by Type B or Type C testing, and the leaks that have been found by Type A tests have been only marginally above existing requirements.
- Given the insensitivity of risk to containment leakage rate and the small fraction of leakage paths detected solely by Type A testing, increasing the interval between integrated leakage-rate tests is possible with minimal impact on public risk. The impact of relaxing the ILRT frequency beyond 1 in 20 years has not been evaluated. Beyond testing the performance of containment penetrations, ILRTs also test integrity of the containment structure.

The findings for CCNPP confirm these general findings on a plant-specific basis considering the severe accidents evaluated for CCNPP, the CCNPP containment failure modes, and the local population surrounding CCNPP.

A. ATTACHMENT 1

PRA Quality Statement for Permanent 15-Year ILRT Extension

The Calvert Cliffs Internal Events and Wind Model, Calvert-CAFTA-TREE-6.2a, was used for this analysis.

An independent PRA peer review was conducted under the auspices of the Pressurized Water Reactor Owners Group in June of 2010, and was performed against the guidance of Regulatory Guide 1.200, Revision 2, and requirements of American Society of Mechanical Engineers (ASME)/American National Standards (ANS) RA-Sa-2009. The scope of the review was a full-scope review of the Calvert Cliffs Nuclear Plant (Calvert Cliffs) at-power, internal initiator PRA.

Findings (generally, documentation issues or model concerns that have been evaluated as not significant using a sensitivity study) have been captured in the PRA Configuration Risk Management Program (CRMP) database. On an on-going basis, other potential PRA model and documentation changes are captured and prioritized in the CRMP database.

To ensure that the current PRA model remains an accurate reflection of the as-built, as-operated plant, the following configuration control activities are routinely performed:

- Design changes and procedure changes are reviewed for their impact on the PRA model. PRA screening is required for all design and procedure changes.
- New engineering calculations and revisions to existing calculations are reviewed for their impact on the PRA model.
- Plant specific initiating event frequencies, failure rates, and maintenance unavailabilities are updated based upon reviews of plant program data, particularly data supporting the Maintenance Rule.

The Calvert Cliffs Internal Events model is also updated to support the Calvert Cliffs Fire PRA.

The Calvert Cliffs Internal Events PRA is based on a detailed model of the plant developed from the Individual Plant Examination for Generic Letter 88-20, "Individual Plant Examination for Severe Accident Vulnerabilities." The model is maintained and updated in accordance with Calvert Cliffs procedures, and has been updated to meet the ASME PRA Standard and Revision 2 of Regulatory Guide 1.200, "An Approach for Determining the Technical Adequacy of Probabilistic Risk Assessment Results for Risk-Informed Activities."

The Calvert Cliffs internal events PRA model was peer reviewed in June 2010. All findings which had significant impact on this analysis have been addressed. This assessment is provided as Table 1. The ILRT application was determined to be an application requiring a Capability Category II PRA model per the Regulatory Guide 1.200 criteria, Revision 2. This is based on the requirement for numerical results for CDF and LERF to determine the risk impact of the requested change and the fact that this change is risk-informed, not risk-based. Table 1 includes discussion of all findings from the industry peer review along with the assessment and evaluation of the finding that shows that they have either been addressed or have no material impact on the ILRT interval extension request.

The peer review found that 97% of the SR's evaluated Met Capability Category II or better. There were 3 SRs that were noted as "not met" and eight that were noted as Category I. As noted in the peer review report, the majority of the findings were documentation related. Of the 11 SRs which did not meet Category II or better, seven were related to conservatism or documentation in LERF and two were related to internal floods. The 3 SRs that were noted as "not met" are LE-F2, LE-G5, and IFQU-A10. LE-F2 relates to LERF results. The dominant LERF contributors were reviewed and model changes implemented prior to the ILRT analysis. LE-G5

relates to the documentation of limitations of applications of the PRA. IFQU-A10 relates to documentation of the treatment of the internal flood analysis in the event trees. None of the "not met" SRs impact the ILRT extension analysis. There were 39 findings. All findings which could be relevant to the ILRT extension evaluation were updated in the internal events model used to quantify the Level 2 release states. Thus, with the exception of minor documentation concerns, the internal events model meets Capability Category II or causes conservative results for all SRs relevant to the ILRT extension evaluation results. No significant changes have been implemented in the internal events PRA. As there are no new methods applied, no follow on or focused peer reviews were required.

The Calvert Cliffs Fire PRA peer review was performed January 16-20, 2012 using the NEI 07-12 Fire PRA peer review process, the ASME PRA Standard (ASME/ANS RA-Sa-2009) and Regulatory Guide 1.200, Rev. 2. The purpose of this review was to establish the technical adequacy of the Fire PRA for the spectrum of potential risk-informed plant licensing applications for which the Fire PRA may be used. The 2012 Calvert Fire PRA peer review was a full-scope review of all of the technical elements of the Calvert Cliffs at-power FPRA (2012 model of record) against all technical elements in Section 4 of the ASME/ANS Combined PRA Standard, including the referenced internal events SRs. The peer review noted a number of facts and observations (F&Os). The findings and their dispositions are provided in Table 2. All findings are being provided and have been dispositioned. All F&Os that were defined as suggestions have been dispositioned and will be available for NRC review. The Fire PRA is adequate to support the ILRT extension.

The Calvert Cliffs seismic PRA model is relatively conservative and, other than the high magnitude acceleration event, is not a dominant contributor. The Calvert Cliffs high winds PRA model is very conservative in the tornado area in that all tornados are grouped into the most conservative event. PRA risk for tornadoes and high winds are based upon IPEEE values. Calvert Cliffs has maintained and updated a high wind PRA model in order to perform risk assessment of tornado missile impacts and hurricane force winds. Although this model has not been peer reviewed in compliance with the ASME/ANS RA-Sa-2009 standard, the model is based upon accepted methodology and utilizes the ASME/ANS RA-Sa-2009 compliant internal events model. High winds updates are not expected to cause a significant increase in CDF or LERF. A more detailed assessment would be expected to cause a decrease in CDF.

Table 1 Internal Events PRA Peer Review – Facts and Observations

F&O ID	SR	Topic	Finding/Observation	Status	Disposition	Impact to ILRT Extension
1-16	AS-B3 SY-B6	Systems Analysis	<p>Based on Sections 2.4 and 2.10 of the System Analysis Introduction Notebook (C0-SY-00, Rev. 0) this SR appears to be met. However, there is a potential issue related to this SR. Did not find reference to any engineering analysis needed to support Containment Air Cooler operation when this system is assumed to be available during LOSP when the containment heats up prior to electrical recovery.</p> <p>(This F&O originated from SR SY-B6)</p>	Complete	<p>The PRA Internal Events Accident Sequence Notebook, C0-AS-001, Section 3.3, has been updated with an engineering analysis of this issue. The analysis identifies that during the Loss of Offsite Power sequences, the Containment Air Coolers are credited for SBO conditions where the containment heats up, and then, after power recovery, the air coolers are credited for containment pressure and temperature control. For these accident sequences, offsite power is restored in one hour, and the containment pressure and corresponding saturation temperature remain well below containment design parameters that would challenge the CACs. Furthermore, failure of CACs is not risk significant, due to the potential availability of containment spray.</p> <p>REFERENCE C0-AS-001</p>	<p>No impact on ILRT analysis. Subsequent analysis has found this issue to be non-significant: 1) the temperature rise is not likely to challenge the containment air coolers, and 2) the importance of the air coolers is significantly reduced by the redundant function provided by containment spray</p>

Table 1 Internal Events PRA Peer Review – Facts and Observations

F&O ID	SR	Topic	Finding/Observation	Status	Disposition	Impact to ILRT Extension
1-17	IFSO-A1 QU-E3	Internal Flooding	Examined Internal Flooding Notebook (C0-IF-001, Rev. 1) Sections 3.0 and 3.1. Part of the Internal Flood analysis may not be complete for assessing the Aux Feedwater Discharge Piping as a Flood Source. (This F&O originated from SR IFSO-A1)	Complete	An engineering analysis has subsequently been performed for AFW discharge piping flooding. The fraction of at-power time during which the AFW system is in operation 0.6% and the AFW Discharge Piping flood may be screened due to their low impact on CDF (<1E-9). REFERENCE C0-IF-001	Due to the relatively low contribution to CDF, this flood has no impact on ILRT analysis.
1-18	IFSO-A4 IFEV-A7	Internal Flooding	Examined Internal Flooding Notebook (C0-IF-001, Rev. 1) Section 3.3 and 5.3. Consideration of human-induced mechanisms as potential flood sources not clear. Regarding human-induced impacts on the flood frequency, Section 5.3 of the IF report states that they were included, but their inclusion should be better documented or referenced from IF (e.g., a sample calculation showing human contribution would be helpful) (This F&O originated from SR IFSO-A4)	Open	Human-induced impacts on the flood initiating event frequencies are not well documented. The issue has been captured in the PRA configuration control database (CRMP), but not yet addressed.	No impact on ILRT analysis. This is a documentation issue.
1-19	IFEV-B3 IFPP-B3 IFQU-B3	Internal Flooding	While some items are included in Section 7.0 of the IF report, many other instances of uncertainties	Open	In the Internal Flood notebook, the discussions on uncertainties and	No impact on ILRT analysis. This is a

Table 1 Internal Events PRA Peer Review – Facts and Observations

F&O ID	SR	Topic	Finding/Observation	Status	Disposition	Impact to ILRT Extension
	IFSN-B3 IFSO-B3		and assumptions are cited throughout the report, but not included in the discussion of Section 7.0 nor are the implications of these other uncertainties and assumptions are discussed. (This F&O originated from SR IFPP-B3)		assumptions should be expanded. This issue has been captured in the PRA configuration control database (CRMP), but not yet addressed.	documentation issue.
1-25	DA-C7	Data	For the most part actual plant-specific data is used as a basis for the number of demands associated actual plant experiences (See basis for DA-C6), which includes both actual, planned and unplanned activities. However, there are a few ESFAS testing and/or other logic channel testing that are not tracked via the plant computer. Created this F&O on non-documentation of ESFAS/logic train testing, which needs to include actual practice. (This F&O originated from SR DA-C7)	Complete	The ESFAS logic train testing has a very low risk significance and generally does not take the logic OOS. The train does go to 2-out-of-3 logic. Occurrences where the train is in 2-out-of-3 logic is incorporated into the PRA Data Analysis Notebook, C0-DA-001, Section 2.6 and 3.5. For the logic relays there is a RAW of <1.04 and Birnbaum on the order of 4E-07. Any logic relay unavailability that does not cause the ESFAS channel to be OOS and bypassed, is therefore of low significance. REFERENCE C0-DA-001	The low risk significance of ESFAS logic train testing is considered to have no impact on ILRT analysis.

Table 1 Internal Events PRA Peer Review – Facts and Observations

F&O ID	SR	Topic	Finding/Observation	Status	Disposition	Impact to ILRT Extension
2-7	IFPP-A5	Internal Flood	<p>Section 2.3 provides a discussion that walkdowns used to confirm plant arrangement. The following note is contained in section 2.3:</p> <p>Unfortunately, the walk-down documentation from the original flooding analysis no longer exists. A plant walk-down was performed as a part of this analysis to provide familiarity with the plant design as well as confirm findings from the original walk-down. This walk-down is documented in a set of notes and photographs included in Appendix B.</p> <p>Walkdown photos for room 105A and 203 show equipment and potential flood propagation paths. However, there is not enough spatial information to develop specific targets for flood impingement or spray.</p> <p>(This F&O originated from SR IFPP-A5)</p>	Complete	<p>A walkdown was performed to assess the susceptibility to jet impingement or spray in rooms 105A and 203. All equipment is considered failed by spray or impingement for flood sources originating in the room. Notebook C0-IF-001 was updated with this additional documentation.</p> <p>REFERENCE C0-IF-001</p>	No impact on ILRT analysis. This is an Internal Flood documentation issue.
2-9	DA-D4	Data	<p>Evidence of meeting this SR at CC-II/III is found in the PRA Data Notebook (C0-DA-001, Rev. 1) in Sections 2.1 and 2.7. Found inconsistencies in the value of total number components of different types (for both units) in Table 2-5</p>	Complete	<p>Table 2-6 of the Data Notebook C0-DA-001 listed incorrect data and Bayesian update results for the SACMs. However, the correct values were used in the models for peer review.</p>	No impact on ILRT analysis. This was a documentation issue. The Internal Events

Table 1 Internal Events PRA Peer Review – Facts and Observations

F&O ID	SR	Topic	Finding/Observation	Status	Disposition	Impact to ILRT Extension
			<p>of the PRA Data Notebook with the actual total number for Calvert Cliffs. Also, found an inconsistency between the prior distribution and posterior distribution for SACM EDG fail to start in Table 2-6 of the Data Notebook.</p> <p>(This F&O originated from SR DA-D4)</p>		<p>For the SACM EDGs in Table 2-6, the correct plant-specific data are in Table 2-5. Table 2-6 lists incorrect data and Bayesian update results for the SACMs. However, the correct values are used in the models.</p> <p>The above errors have been corrected in C0-DA-001. Other minor typographical errors were identified and corrected in the notebook.</p> <p>REFERENCE C0-DA-001</p>	<p>model includes the correct data.</p>
3-3	SY-C2	Systems Analysis	<p>Section 2.3 of each system notebook states that marked up plant system drawings are provided as supplements to the system notebook, which depicts the boundary of the system in terms of PRA modeling. The drawings are not in the notebooks.</p> <p>(This F&O originated from SR SY-C2)</p>	Complete	<p>Marked-up system boundary drawings were generated for each system notebook. Where Unit 1 and Unit 2 are similar, just the Unit 1 boundary is depicted. In addition, the system notebooks include drawing snippets, sketches, and descriptive text that also depict the system boundary.</p> <p>REFERENCES C0-SY-[All]</p>	<p>No impact on ILRT analysis. This is an Internal Events documentation issue that has been addressed.</p>

Table 1 Internal Events PRA Peer Review – Facts and Observations

F&O ID	SR	Topic	Finding/Observation	Status	Disposition	Impact to ILRT Extension
3-5	SY-A11 SY-A6	Systems Analysis	<p>The fault tree does not include potential failures of the AFW accumulator system.</p> <p>(This F&O originated from SR SY-A11)</p>	Complete	<p>A bounding sensitivity case was run to include failure of the AFW accumulators failing short-term AFW operation. This issue has an insignificant contribution to CDF. Short-term failure of the AFW operation is dominated by failure of electrical support systems and failure of active hardware (i.e. valves and instrumentation). The applicable system notebooks were updated.</p> <p>REFERENCES C0-SY-036 C0-SY-019 C0-SY-000</p>	<p>This finding does not impact the ILRT extension. The random failure probability of the accumulators is two orders of magnitude lower than active hardware failures that support the same system function.</p>
3-8	SY-C1 SY-A13	Systems Analysis	<p>Several system notebooks were reviewed (AFW, EDG, SI, 120 VAC electrical, etc.). In general, the documentation is complete and thorough. In most cases it clearly follows the RG 1.200 SRs. In some places, assumptions were imbedded in the documentation without sufficient reference or justification. Examples include:</p> <p>SI notebook page 11, last bullet 'Only one of the three HPSI pumps functions - For a cold leg break, it</p>	Complete	<p>Some new flow diversions were identified as part of the Fire PRA Multiple Spurious Operation review, and these were added to the system models and system notebooks. Furthermore, a comprehensive review of PRA mechanical systems notebooks and drawings was performed to identify and document potential flow diversions. Flow diversion discussions were added to</p>	<p>No impact on ILRT analysis. This is an Internal Events documentation issue that has been addressed.</p>

Table 1 Internal Events PRA Peer Review – Facts and Observations

F&O ID	SR	Topic	Finding/Observation	Status	Disposition	Impact to ILRT Extension
			<p>is assumed that only one-fourth pump discharge is spilled via the break. For a hot leg break, the entire pump discharge reaches the core.'</p>		<p>Sections 3.4.d of the applicable system notebooks.</p>	
			<p>SI notebook page 12, 2nd bullet 'The maximum time assumed for operation for the safety injection pumps is 30 seconds following SIAS initiation.' C0-SY-000 states that each system notebook addresses flow diversions (where applicable) in section 3.4.d. Although flow diversions appear to be addressed (for example, the SW notebook talks about flow diversion), there is no consistent discussion in each system notebook. (This F&O originated from SR SY-C1)</p>			
3-9	DA-B1	Data	<p>DA notebook table 2-5 contains the grouping of components for plant specific failure data. Many of the groupings appear to take into account differences in such things as size, type, mission type (e.g., FW TDP run vs. AFW TDP standby). However, in some cases, it is not clear what the basis for the grouping is. For example, SW MDP RUN and SRW MDP RUN are grouped together even though</p>	Complete	<p>The model has been updated to add additional component types and failure modes to better reflect service conditions. Service Water and Salt Water pumps were broken out. AFW pumps and Safety Injection pumps were broken out. This resulted in changes to the associated failure rates. The change has been</p>	<p>No impact on ILRT analysis. The model used for the ILRT analysis includes the updated data and failure modes.</p>

Table 1 Internal Events PRA Peer Review – Facts and Observations

F&O ID	SR	Topic	Finding/Observation	Status	Disposition	Impact to ILRT Extension
			<p>they are of different service conditions (salt water vs. clean water), voltages (480 VAC vs. 4160 VAC), size, etc. Similarly, AFW MDP is included with HPSI MDP and LPSI MDP, even though the two SI pumps are pumping borated water, while the AFW pump is pumping condensate grade water. No documentation of the appropriateness of these groupings is provided.</p>		<p>reflected in the Data Notebook, C0-DA-001. REFERENCE C0-DA-001</p>	
			<p>(This F&O originated from SR DA-B1)</p>			
3-11	QU-B7	Quantification	<p>The mutually exclusive cutsets for each system are described in the system notebook section 3.4.e. Several SY notebooks were reviewed to determine appropriateness of the mutually exclusive cutsets. All appeared reasonable. A review was performed of the MUTEX gate within the fault tree model and the appropriate combinations identified in the SY notebooks appear to have been included in the model. There are two gates under the MUTEX gate which contain mutually exclusive cutsets which are not documented in the system notebooks. While the majority of these are intuitively obvious (e.g.,</p>	Complete	<p>A comprehensive review of mutual exclusive modeling was performed. Each system notebook and each system model was reviewed to validate the appropriateness of the modeling and reconcile any differences, and to verify that a documented basis exists for each mutually exclusive event. The PRA model was updated to reflect new, deleted, or re-organized mutually exclusive modeling identified as part of this review.</p>	<p>No impact on ILRT analysis. The PRA model that was updated as part of this review was used as the model for the ILRT analysis.</p>

Table 1 Internal Events PRA Peer Review – Facts and Observations

F&O ID	SR	Topic	Finding/Observation	Status	Disposition	Impact to ILRT Extension
			11 Steam Generator Tube Rupture occurs as an IE AND 12 Steam Generator Tube Rupture occurs as an IE), these should be included in an appropriate system notebook. (This F&O originated from SR QU-B7)		REFERENCE SY-C0-[ALL]	
3-12	QU-D3	Quantification	A review of the top cutsets from each event tree was performed. The utility stated that during this review, cutsets were reviewed to determine if any mutually exclusive events were contained within cutsets, if any flag settings were inappropriate or if any recoveries were overlooked or added inappropriately. A review of a sampling of cutsets did not indicate any inappropriate results. However, the QU notebook does not include a discussion of this review. (This F&O originated from SR QU-D3)	Complete	Documentation of the cutset reviews was presented to the peer review team; although, the documentation was separate from the formal QU notebook package. A note was added to the QU notebook directing the reader to the location of the cutset review notes and spreadsheets. The PRA configuration control procedure, CNG-CM-1.01-3003, requires a review of cutsets for PRA changes. In practice, the top CDF and LERF cutsets are examined for even the most innocuous model changes. REFERENCE CNG-CM-1.01-3003 C0-QU-001 C0-FRQ-001	No impact on ILRT analysis. The original internal events cutset review notes have now been archived.
4-5	IE-A10 SY-A10	Initiating Events	The only mention in C0-SC-001 of shared systems between the units	Complete	To address this finding, the Diesel Generator modeling	No impact on ILRT analysis.

Table 1 Internal Events PRA Peer Review – Facts and Observations

F&O ID	SR	Topic	Finding/Observation	Status	Disposition	Impact to ILRT Extension
	IE-C3 SC-A4		<p>is the SBO EDG, noted in Section 4.1.2. It states that the SBO diesel can power any one bus on either unit. However, in the CAFTA model, there is an assumed bus preference of 11, then 24, then 12, then 23.* This is noted in the EDG system notebook but no basis is provided. The procedures do not actually have a preference, which yields a potentially non-conservative analysis. For example, if there is a LOOP, the U2 diesels fail to start and the U1 diesels fail to run after 1 hour. The SBO diesel would then be aligned to U2, and it is non-conservative to give the U1 bus 11 full credit. If such non-conservatism is negligible, some analysis should be performed to demonstrate this.</p> <p>(This F&O originated from SR IE-A10)</p> <p>*Note: Peer review finding was not precise. It should have stated bus preference for Unit 1 is 11, then 24, and for Unit 2, is 24 then 21.</p>		<p>was updated as described in Appendix H of C0-SY-023-024, PRA DG System Notebook. EOP-7 directs to align the 0C DG to the unit with redundant safety equipment out-of-service, with a goal to restore at least one 4KV bus. Since 4KV Buses 11 and 24 support AFW, those busses would have a preference over Busses 14 and 21, all else being equal. No unit preference is modeled. If there is a conflict in the order-of-preference, for example, both 4KV Bus 11 and 4KV Bus 24 are not powered, then a 50-50 probability is assumed as to the preferred bus.</p> <p>REFERENCE C0-SY-023-024</p>	<p>The finding has been addressed in the Internal Events model, which, in turn, is used in the ILRT analysis.</p>
4-12	HR-C1	Human Reliability	<p>One basic event calculated in the appendix (ESF0HFCISZFG) was not included in the fault tree models. CCNPP staff noted that it</p>	Complete	<p>The basic event has been added to the model. A sensitivity run with the basic event included in the current</p>	<p>No impact on ILRT analysis. The missing basic event has</p>

Table 1 Internal Events PRA Peer Review – Facts and Observations

F&O ID	SR	Topic	Finding/Observation	Status	Disposition	Impact to ILRT Extension
			had previously been modeled, but inadvertently deleted in an update. (This F&O originated from SR HR-C1)		model showed no increase in risk. The system notebook C0-SY-048 was updated. REFERENCE C0-SY-048	been added to the internal events model used in the ILRT analysis.
4-15	IFEV-A6	Internal Flooding	The internal flooding analysis did not have a formal process to gather plant specific design information, operating practices, etc. that could potentially affect the generic flooding frequencies. In response to an NRC RAI on the CCNPP ISI program plan, CCNPP mentioned a review of Condition Reports that did not find any items that would increase the flooding frequency. The CR review meets part of the requirement, but the SR also calls for reviews of plant design, operating practices, etc. that should be considered. The evaluation should be documented in the PRA. (This F&O originated from SR IFEV-A6)	Open	This finding has been identified in the PRA configuration control database (CRMP), but has not yet been addressed.	No impact on ILRT analysis. The review of condition reports did not identify any design issues or operating practices that would affect the generic flooding frequencies.

Table 1 Internal Events PRA Peer Review – Facts and Observations

F&O ID	SR	Topic	Finding/Observation	Status	Disposition	Impact to ILRT Extension
4-19	LE-C13 LE-F3 LE-G4	Large Early Release	<p>The sources of uncertainty are well identified in Table 5-1 of the LE notebook and quantified in Table 5-2 of the QU notebook. However, no discussion of the uncertainties or insights from them is provided. For example, Sensitivity 1 shows a 74% reduction in LERF, but this large reduction is not investigated.</p> <p>Also, conservatisms in the ISLOCA analyses were discussed in the AS review. SGTR was treated in an overly conservative manner by categorizing all SGTR as LERF.</p> <p>(This F&O originated from SR LE-F3)</p>	Complete	<p>Dominant LERF cutsets were reviewed to identify uncertainties that could be addressed. Two changes have been implemented to address significant uncertainties and reduced LERF. First, a reverse-flow check valve in the CVCS Letdown line was credited as a potential ISLOCA recovery. Second, a new human action was added with realistic timing for Steam Generator isolation and RCS depressurization on a SGTR. These and less significant model updates resulted in a LERF-to-CDF ratio change from approximately 17% to approximately 10%. This newer ratio is in the typical range for other PWRs.</p> <p>REFERENCE C0-LE-001</p>	<p>No significant impact on ILRT analysis. The dominant LERF contributors were reviewed and model changes implemented. The Calvert Cliffs LERF contribution is now similar to other PWRs.</p>
4-20	LE-F1 LE-G3	Large Early Release	<p>The relative contribution to LERF is presented in the QU notebook by PDS and by initiating event, but not by accident progression sequence, phenomena, containment challenges or containment failure mode.</p>	Complete	<p>The contributions to LERF are documented in the Quantification Notebook and are noted as such in the Level 2 Notebook. Accident progression sequences are located in Section 4.2.3 and</p>	<p>No impact on ILRT analysis. This is an internal events documentation issue.</p>

Table 1 Internal Events PRA Peer Review – Facts and Observations

F&O ID	SR	Topic	Finding/Observation	Status	Disposition	Impact to ILRT Extension
			(This F&O originated from SR LE-G3)		Appendix C. The Level 2 notebook has been updated to point to additional phenomena and containment challenges and failure mode Tables/Figures in the QU Notebook REFERENCE C0-QU-001 C0-LE-001	
4-21	LE-G5	Large Early Release	The LE notebook states that limitations in the LE analysis that could impact applications are documented in the QU notebook, but it is not. Therefore, SR LE-G5 was determined to be not met. Given the conservative modeling of SGTR and ISLOCA, the impact on applications should include assessment of how this conservatism can skew the LERF results. (This F&O originated from SR LE-G5)	Complete	Section 5.5.2.7 of C0-LE-001, Revision 2 - added discussion of results of impact on application of the Unit 2 ILRT extension request. REFERENCE C0-LE-001	This internal events finding does not impact the ILRT analysis.
4-22	LE-C10 LE-C12 LE-F2 LE-C3	Large Early Release	The LERF contributors have not been reviewed for reasonableness (per SR LE-F2). Therefore, SR LE-F2 was determined to be not met. The QU notebook discusses the top 20 LERF cutsets (which total	Complete	The LERF results were reviewed for conservatisms as described in the SRs. After conservatisms were addressed (see discussion for F&O 4-19 above), no	No significant impact on ILRT analysis. The dominant LERF contributors were reviewed and

Table 1 Internal Events PRA Peer Review – Facts and Observations

F&O ID	SR	Topic	Finding/Observation	Status	Disposition	Impact to ILRT Extension
			<p>73% of the total LERF). It notes conservatism in the cutsets and says it will be evaluated in Section 5.2, but is not. Section 4.3.6 of the QU notebook compares the total LERF of CCNPP to St. Lucie, but does not even break the results down by contributor (e.g., SGTR, ISLOCA, etc.).</p> <p>Also, the ASME PRA Standard SRs C-3, C-10 and C-13 require a review of the LERF results for conservatism in the following areas:</p> <ol style="list-style-type: none"> 1. Engineering analyses to support continued equipment operation or operator actions during severe accident progression that could reduce the LERF 2. Engineering analyses to support continued equipment operation or operations after containment failure. 3. Potential credit for repair of equipment. <p>No such review has been performed, despite the large conservatism noted in the containment bypasses. (This F&O originated from SR LE-F2)</p>		<p>significant issues were identified.</p> <p>REFERENCE C0-LE-001</p>	<p>model changes implemented. The Calvert Cliffs LERF contribution is now similar to other PWRs.</p>
5-10	LE-D7	Large Early Release	Following the failure of one or more containment penetrations to isolate on CIAS, a feasible	Complete	The merits have been considered of adding an operator action in order	No impact to ILRT analysis. Modeling of an

Table 1 Internal Events PRA Peer Review – Facts and Observations

F&O ID	SR	Topic	Finding/Observation	Status	Disposition	Impact to ILRT Extension
			<p>operator action is to manually close the failed valves from the Main Control Room.</p> <p>(This F&O originated from SR LE-D7)</p>		<p>close containment penetration from the Main Control Room to recover from a containment isolation failure. A review of cutsets shows that a recovery is not feasible for top LERF sequences, because the sequence includes either 1) a loss of CR indication, 2) includes a station black-out condition, or 3) includes non-recoverable pipe breaks.</p> <p>REFERENCE C0-LE-001 Attachment S</p>	<p>operator action to manually close failed valves from the main control room would not significantly reduce LERF, as such an action is not feasible for the significant sequences where containment isolation has failed.</p>
5-17	IE-C1 IE-C13 IE-C4	Initiating Events	<p>Bayesian updates of non-time-based LOCA data were improper. The small and medium LOCA frequencies were obtained from draft NUREG 1829 then Bayesian updated (in App E) with CCNPP experience from 2004 to 2008. The Very Small LOCA prior having alpha = 0.4, Mean = 1.57E-03; was Bayesian updated to a Posterior having a mean value of 7.02E-04. This represents an excessive drop associated with CCNPP experience of 4 to 5 years. Similarly, the Small and Medium LOCAs were Bayesian updated</p>	Complete	<p>CENG understands the general concern on Bayesian updating of rare events. However, the method used was based on a white paper developed by industry experts regarding LOCA frequencies. These experts included INL, NRC and Industry experts. In addition, the approach used for the Calvert PRA was the same as used for the NRC SPAR model. This issue is captured in the PRA configuration control</p>	<p>No impact on ILRT analysis. The approach used for LOCA frequencies has been validated by industry experts and is the same approach as was used for the NRC's SPAR model.</p>

Table 1 Internal Events PRA Peer Review – Facts and Observations

F&O ID	SR	Topic	Finding/Observation	Status	Disposition	Impact to ILRT Extension
			<p>with the whole industry experience rcy data. The draft NUREG 1829 LOCA frequencies were obtained from expert elicitations (not time-based) that included crack propagation analysis. The Bayesian update for VSLOCA used the Alpha parameter and the mean value to justify that the prior mean was based on 255 rcy. This may not have been the basis for the expert elicitations in NUREG 1829.</p> <p>Also, the Medium LOCA frequency may be classified as extremely rare event. It would require no Bayesian updating. The current CCNPP SLOCA and MLOCA frequencies are very close even though the source data in NUREG 1829 indicates a negative exponential drop in these frequencies.</p> <p>(This F&O originated from SR IE-C1) (Note: rcy – reactor year)</p>		<p>database (CRMP).</p> <p>REFERENCE C0-IE-001</p>	
5-18	IE-C2 IE-C7	Initiating Events	<p>Justify the exclusion of LOOP event at CCNPP in 1987. No time trend analysis was provided to justify the exclusion.</p> <p>(This F&O originated from SR IE-C2)</p>	Complete	<p>The event is not counted following guidance provided in NUREG/CR-6928, based upon trend analysis. A full discussion is included in the Initiating Event notebook, C0-IE-001.</p>	<p>No impact on ILRT analysis. The data analysis is acceptable.</p>

Table 1 Internal Events PRA Peer Review – Facts and Observations

F&O ID	SR	Topic	Finding/Observation	Status	Disposition	Impact to ILRT Extension
					REFERENCE C0-IE-001	
5-23	HR-A2	Human Reliability	<p>The Pre-Initiator HRAs did not include the miscalibration of SIT pressure. For example, in the event where SIT pressure is miscalibrated high, various accident scenarios requiring SI are negatively impacted. Add SIT pressure miscalibrated high or, justify no impact on CDF / LERF.</p> <p>(This F&O originated from SR HR-A2)</p>	Complete	<p>It is agreed that the miscalibration of SIT pressure could have a negative impact on various accident scenarios involving LLOCA and VLLOCA initiators. However, this instrumentation is not modeled explicitly and is therefore deemed included within the component boundary for the SIT. As such the miscalibration probability would be included in the SIT unavailability.</p> <p>REFERENCE C0-HR-001</p>	<p>No impact on ILRT analysis. Given the pressure of the CCNPP SITs they are only required and provide significant benefit on Large LOCAs. The frequency of a Large LOCA times the pre-initiator frequency is negligible.</p>
5-25	SC-C1 HR-I2 SC-C2	Success Criteria	<p>Simplify the traceability of Tsw. In the post initiator HRA details, the HRA success criteria are often provided as a positive re-statement of the HRA title. And, the consequence of failure is often stated as core damage. Consider adding Tsw to the success criteria and linking that to the PCTran case where Tsw was developed.</p>	Complete	<p>Where applicable, the Tsw of each HFE that could be traced to the Success Criteria notebook (CS-SC-001) was updated and referenced in the HRA Calculator. C0-HR-001 was also updated.</p>	<p>No impact on ILRT analysis. This is an internal events documentation finding.</p>

Table 1 Internal Events PRA Peer Review – Facts and Observations

F&O ID	SR	Topic	Finding/Observation	Status	Disposition	Impact to ILRT Extension
			Also, in the SC report (Table B-3), consider adding the actual time to core uncover (or core damage) instead of providing a "Yes" entry in the column of "core damage?" (This F&O originated from SR HR-12)		REFERENCE C0-HR-001	
5-30	LE-D1 LE-B2	Large Early Release	Section 3.2.11 discussed the containment challenge from Hydrogen Combustion. It concluded that the challenge may be significant for some accident scenarios. The CCNP entry in Table 6.11-2 of the Level 2 WCAP showed a potentially significant impact from Hydrogen burn. Provide an estimate of the impact of Hydrogen burn on containment pressure. Use an accident scenario that is likely to produce larger amounts of H ₂ with failed containment spray. The optimal time to estimate the impact of Hydrogen burn is approximately at 2 hours which is the time when the EOF and TSC personnel have convened and are ready to guide the Main Control Room into periodic Hydrogen burns before the formation of explosive mixtures.	Complete	CCNPP's Level 2 PRA follows the analysis in WCAP-16341-P, Simplified Level 2 Modeling Guidelines. In the industry-supported analysis, the percentage of cladding oxidation is the main factor used to develop a maximum H ₂ concentration in the containment, and, in turn, a containment pressure is calculated if the H ₂ completely burns. These are then mapped to site-specific containment failure probabilities. A simplifying assumption is made that "no pre-burning of hydrogen generated in the core melt progression is considered." Calvert Cliffs' severe action management procedures do include	No impact on ILRT analysis. The methodology in WCAP-16341-P is appropriate for Calvert Cliffs level 2 analysis for internal events initiators.

Table 1 Internal Events PRA Peer Review – Facts and Observations

F&O ID	SR	Topic	Finding/Observation	Status	Disposition	Impact to ILRT Extension
			(This F&O originated from SR LE-D1)		actions to reduce H ₂ concentration in the containment, but these actions are not credited in the PRA model. Also, Containment Spray is not questioned for the LERF accident sequences. Containment Spray is a factor in LATE containment failure accident sequences.	
					REFERENCES C0-LE-001	
5-31	DA-D4	Data	The summary table for Bayesian updated parameters (on Page 53 of the PRA Data Notebook, C0-DA-001, Rev. 1) shows the CS-MDP was Bayesian updated with plant experience containing 1 failure and Zero run-hours. The CCNPP PRA staff responded to this issue as an isolated case. There is an actual FTR > 1 hr	Complete	The aforementioned footnote was incorporated into Table 2-6 of C0-DA-001. REFERENCE C0-DA-001	No impact on the ILRT analysis for this minor internal events documentation issue and no changes were required for the CS-MDP failure rate.
			(This F&O originated from SR DA-D4)			
6-3	SC-B2	Success Criteria	Expert judgment was not used as the sole basis for any success criteria. However, upon inspection of the PCTran run tables in the SC report appendices, many instances	Complete	The approach for SLOCA break size analysis is discussed in the Success Criteria notebook. Furthermore, a review was	The existing analysis meets the intent of the SR and therefore there is no

Table 1 Internal Events PRA Peer Review – Facts and Observations

F&O ID	SR	Topic	Finding/Observation	Status	Disposition	Impact to ILRT Extension
			<p>of surrogate or inferred results were found. Instead of running specific PCTran calculations to cover the whole SLOCA break size spectrum, intermediate break sizes have been calculated supplemented with expert judgment to derive limiting time delay for operators to actuate SI (30 min) or limiting time delay for OTCC (SGL<350'+10min).</p>		<p>conducted of this issue; in addition, TH analyses were completed to verify the break-size ranges. It was found that the computer simulations adequately represented the various break-size ranges.</p>	<p>impact on the ILRT analysis.</p>
			<p>(This F&O originated from SR SC-B2)</p>		<p>REFERENCE C0-SC-001</p>	
6-5	SY-A20	Systems Analysis	<p>When appropriate, the simultaneous unavailability within a system is documented in the system notebooks and included in the PRA model. However, a further review of these items is required for completeness.</p>	Complete	<p>AFW basic event AFW0TMMMAINT6-F7 was determined to not be needed in the plant model. The basic event was removed. All remaining AFW equipment unavailability events in the model and notebooks were reviewed for consistency. AFW0TMMMAINT-TF was determined to be modeled correctly, its description was found to be in error in the system notebook. Notebook C0-SY-036 was updated. A review for concurrent maintenance was previously performed and documented in the Data Notebook.</p>	<p>No impact on ILRT analysis. The offending basic event was removed from the model. A review did not discover other missing or incorrect simultaneous unavailability events.</p>
			<p>(This F&O originated from SR SY-A20)</p>			

Table 1 Internal Events PRA Peer Review – Facts and Observations

F&O ID	SR	Topic	Finding/Observation	Status	Disposition	Impact to ILRT Extension
					REFERENCE C0-SY-036	
6-8	HR-H2	Human Reliability	<p>Some recovery actions included in the model (thus credited) are set to screening values. In the HEP evaluation (appendices of the HR report) there are no indications that procedures, training, or other shaping factors are available on a plant-specific basis.</p> <p>(This F&O originated from SR HR-H2)</p>	Complete	<p>For each screening HRA, the internal events analysis was updated to include a specific reference to the earlier HRA analysis. Included are the applicable success criteria for each recovery. Refer to C0-HR-001, Internal Events Human Reliability Analysis, and the associated HRA Calculator file.</p> <p>For Fire PRA development, the internal events HRAs with screening values were analyzed to assure that they were sufficiently conservative for fire scenarios. Refer to Section 4.1 of C0-HRA-001, Fire PRA Human Reliability Analysis. Documentation for fire HRA actions are similar</p>	No impact on ILRT analysis. The documentation for internal events HRAs was updated to address this finding.

Table 1 Internal Events PRA Peer Review – Facts and Observations

F&O ID	SR	Topic	Finding/Observation	Status	Disposition	Impact to ILRT Extension
					to that done for the updated internal events HRA actions. REFERENCE C0-HR-001 C0-HRA-001	
6-9	HR-11	Human Reliability	The HR report is well documented in general and will facilitate upgrades, however, some basic event names are not consistent between the HR report and the system notebooks. (This F&O originated from SR HR-11)	Complete	Updated the notebooks in the reference section so HRA designator names and descriptions are the same in the HR Calculator, HR notebook, CAFTA Model 6.0. Changes included adding the "-B" extension and removing the "(-2)" event where applicable. REFERENCE C0-HR-001 C0-SY-[Many]	No impact on Fire PRA. This is a documentation finding. HRA names in the model and notebook are now consistent.
6-10	IFPP-A2 IFSN-A2	Internal Flooding	Plant design features such as open rooms or as built divisions are used to define the flood areas and was well documented. More detail is needed as to why the containment buildings were screened from the analysis.	Complete	The Internal Flood notebook has been updated to incorporate an analysis describing the screening of the containment building from flooding analysis. Essentially, the containment is designed for LOCA condition, which screens	No impact on ILRT analysis. This is a documentation finding for the Internal Flood notebook.

Table 1 Internal Events PRA Peer Review – Facts and Observations

F&O ID	SR	Topic	Finding/Observation	Status	Disposition	Impact to ILRT Extension
			(This F&O originated from SR IFPP-A2)		reactor coolant system and related piping system. Other piping systems have limited inventory, are normally isolated, or have a low flow rate. Reference C0-IF-001. REFERENCE C0-IF-001	
6-14	IFSO-B1 IFSN-A9	Internal Flooding	While the flooding calculations have been performed and are thought to be correct and well done, additional documentation of data would enhance the IF report. It appears that the input reports and references are based on poorly documented or non-officially revised reports and information sources. (This F&O originated from SR IFSN-A9)	Open	This is a documentation finding for the internal floods notebook. The issue has been captured in the PRA configuration control database (CRMP), but not yet closed-out.	No impact on ILRT analysis. This is a documentation issue.
6-16	IFQU-A11 IFPP-B2	Internal Flooding	Walkdowns have been conducted and are documented in Appendix B of the IF report. It is stated in the IF report that prior information is no longer available; this fact should be corrected as required for analysis updates and information verifications.	Open	This is an internal floods documentation finding. The finding has been captured in the PRA configuration control database (CRMP), but not yet addressed.	No impact on ILRT analysis. This is a documentation issue.

Table 1 Internal Events PRA Peer Review – Facts and Observations

F&O ID	SR	Topic	Finding/Observation	Status	Disposition	Impact to ILRT Extension
			(This F&O originated from SR IFQU-A11)			
6-17	IFQU-A10	Internal Flooding	<p>By including the flooding events under the transient fault tree, the LERF impacts are automatically accounted for in the same manner as the general transient events in the LERF analysis. Very little documentation is found related to the IF analysis in the LE report, although the IF report states that the LERF impacts due to flooding are documented and analyzed in the LE report. Therefore, SR IFQU-A10 was determined to be not met.</p> <p>(This F&O originated from SR IFQU-A10)</p>	Open	<p>The level of modeling detail in the CCPRA is sufficiently robust such that the model logic for flood impacts propagate appropriately through the system fault trees so that the equivalent general transient initiator (e.g loss of CCW) is appropriately defined in the transient fault tree. In addition, cutset reviews have not revealed the current modeling to be deficient in this regard. This documentation of the above basis is captured in the PRA configuration control database (CRMP), but not yet addressed.</p>	<p>No impact on ILRT analysis. This is a documentation issue.</p>
6-18	HR-H2	Human Reliability	<p>The system time window Tsw for post initiator HRAs was frequently associated with 'core damage'. Post initiator HRAs that appear in the top cutsets may require success criteria linked to beginning of core uncover (about 20 minutes before 'core damage'). Or, the operator actions that may fall into that final 20-minute time</p>	Complete	<p>It was determined that the text in Section 3.1.5.7 was incorrect and does not capture how stress is actually applied in the EPRI HRA Calculator. C0-HR-001, Internal Events PRA Human Reliability Analysis, has been updated to show the stress level applied to each</p>	<p>No impact on ILRT analysis. As described in this F&O for internal events, the stress levels in the model are appropriate, but updates to the documentation</p>

Table 1 Internal Events PRA Peer Review – Facts and Observations

F&O ID	SR	Topic	Finding/Observation	Status	Disposition	Impact to ILRT Extension
			<p>period should be overridden to assume a high stress level. While section 3.1.5.7 described this approach, there is no evidence of its proper application in the HRA quantifications.</p>		<p>HFE and the justification for stress selection. Also included is a correlation between stress level and failure of execution probability. New text has been provided for inclusion in a future update of the HRA notebook.</p>	<p>are required. The internal events documentation was updated.</p>
			<p>(This F&O originated from SR HR-H2)</p>		<p>For the Fire PRA development, the internal events HRA stress levels were carried forward. As described in C0-HRA-001, Fire PRA Human Reliability Analysis, additional stresses were evaluated and incorporated due to the fire initiator.</p>	
					<p>REFERENCE C0-HR-001 C0-HRA-001</p>	
6-22	HR-E1	Human Reliability	<p>Upon RAS, LPSI stops and EOP-5, Step S.1(d) requires the Operators to 'Shut RWT OUT Valves SI-4142, 4143'. This manual action was not modeled in the PRA. The CCNPP PRA staff provided reasonable response to this issue. Based on CR-2009-005581, there is no impact on</p>	Complete	<p>As documented in CR-2009-005881, shutting the RWT outlet valves upon a RAS does not impact station operability. The Safety Injection Pumps and Containment Spray Pumps will not fail if the RWT isolation valves do not close</p>	<p>No impact on ILRT analysis. The system is operable without the manual action to shut the RWT outlet valves. There is no impact on</p>

Table 1 Internal Events PRA Peer Review – Facts and Observations

F&O ID	SR	Topic	Finding/Observation	Status	Disposition	Impact to ILRT Extension
			<p>pump operability. Also, the staff will continue to track the CR. If there are any changes to the disposition of pump operability, then a new HRA may be added to the PRA model (if warranted).</p> <p>(This F&O originated from SR HR-E1)</p>		<p>with a RAS signal. A design margin issue has been identified. This issue has been added to the plant's margin management program. No model changes have been made, but the PRA configuration management program, CNG-CM-1.01-3003, would capture any design changes concerning this issue.</p> <p>REFERENCE C0-SY-052 CR-2009-005881 CNG-CM-1.01-3003</p>	<p>internal events CDF. The issue was added to the plant's margin management program.</p>
6-23	HR-G7	Human Reliability	<p>When the Calculator reads in the combinations, it assumes that actions occur in the order of the time delay (Td). However, the time delay is not the same for all sequences, and care must be taken to make the combinations appropriate for the sequences in which they occur. Page 88 of the HRA notebook indicates this was considered, since the Td was modified for events occurring prior to reactor trip, and also for OTCC after SG overfill. However, not all occurrences have been addressed. The combination</p>	Open	<p>New HRA events, CVC0HFBHEOTA-B-8HRS and AFW0HF-CC-SGDEC-8HR were added to model Td variances where CST depletion occurs early and when it occurs later. This accounts for appropriate sequencing of events.</p> <p>This specific issue with time delay and CST depletion has been addressed and incorporated into the PRA model. An updated dependency analysis has</p>	<p>No impact on ILRT analysis. The new HRA events are not significant.</p>

Table 1 Internal Events PRA Peer Review – Facts and Observations

F&O ID	SR	Topic	Finding/Observation	Status	Disposition	Impact to ILRT Extension
			<p>examined by the review team is Combination 770 (OTCC after CST depletion). In this event the CST depletion should come first.</p> <p>(This F&O originated from SR HR-G7)</p>		<p>been performed, which includes these new HRA events. The dependency analysis shows that these new HRA actions are not significant for CDF or LERF. A PRA configuration control database (CRMP) item has been initiated to formally incorporate the updated dependency analysis into the model.</p> <p>REFERENCE C0-HR-001 C0-HRA-001</p>	
7-13	QU-A2	Quantification	<p>Discrepancy between documentation and result files. SBO037 and SBO038 sequences appear to be inverted in Tables D-1, 4.2.2, 4.2.4, 4.2.5, B-3).</p> <p>(This F&O originated from SR QU-A2)</p>	Complete	<p>The top flood cutset was incorrectly flagged as being SBO sequence 37 (offsite power recovered < 1 hour) instead of sequence 38 (offsite power not recovered). Updated tables B-2, C-1, and D 1 in C0-QU-001. Spot-check was performed to identify other errors. In C0-QU-002, fixed sequence 12 table 4.2-5, which incorrectly showed sequence 37 instead of 38.</p>	<p>No impact on ILRT analysis for this internal events documentation issue.</p>

Table 1 Internal Events PRA Peer Review – Facts and Observations

F&O ID	SR	Topic	Finding/Observation	Status	Disposition	Impact to ILRT Extension
					REFERENCE C0-QU-001 C0-QU-002	

Table 2 Fire PRA Peer Review – Facts and Observations – Findings

F&O ID	SR	Topic	Status	Finding	Disposition	Impact on ILRT Analysis
PP-B3-01	PP-B3 PP-B6 PP-C3	Plant Partitioning	Complete	<p>The containment is partitioned into 2 PAUs. There are intervening combustibles and this was accounted for in the PRA by treating the 20 feet as an overlap region and failing components affected in both PAUs. There is no justification given for the 20 foot assumption. The turbine deck is continuous from unit 1 to unit 2. This area is divided into 2 PAUs, TURB1 and TURB2, but there is no discussion for the basis of the partitioning. Finding level of significance is based on crediting spatial separation with no requisite justification.</p> <p>Maintain the containment as 1 PAU and discern the separation of east from west in the fire modeling. Document the spatial separation and no intervening combustibles for the turbine deck.</p>	<p>C0-PP-001, Calvert Cliffs Fire PRA Plant Partitioning Notebook, was updated to include an analysis that justifies the partitioning of the containment into two plant partitioning units with a 20-foot spatial separation (known as the buffer zone). The only potential intervening combustibles in this buffer zone were identified as qualified cables that were verified to be encased within marinate covered raceways. The covers prevent the cables from becoming potential combustibles and therefore are not considered intervening combustibles.</p> <p>The unit 1 and unit 2 Turbine Deck was walked down to assess for the acceptability of the Appendix R partitioning into distinct PAUs. The boundary was assessed to have at least a 20-foot separation between potential ignition sources and potential targets, assessed for intervening combustibles, and the Turbine deck volume assessed for damaging hot gas layer development. The partitioning was found acceptable and consistent with NUREG/CR-6850, Section 1.5.2, where main turbine decks are typical applications where spatial separation has been credited.</p>	No impact to ILRT analysis, as this affects the FPRA plant partitioning analysis.
PP-B5-01	PP-B5 PP-C3	Plant Partitioning	Complete	The water curtain in the CCW room was credited as an active fire barrier. The justification	The Component Cooling Water room water curtain is an approved Appendix R exemption, as identified in the exemption	No impact to ILRT analysis, as this affects

Table 2 Fire PRA Peer Review – Facts and Observations – Findings

F&O ID	SR	Topic	Status	Finding	Disposition	Impact on ILRT Analysis
				<p>was that the water curtain was part of the original regulatory fire protection program. This meets CAT 1, but needs enhancement for CAT II/III. Finding level was used because the requirements for CAT II/III were not met.</p> <p>Calvert Cliffs should provide a direct reference to their Appendix R program as the basis for the acceptability for this or provide a design basis justification for the water curtain and document that in the PP notebook if the Appendix R program reference cannot be found.</p>	<p>issued by the NRC in response to Calvert Cliffs exemption request ER820816. The validity of crediting CCW Room Water Curtains is discussed in Southwest Research Institute Report No. 01-0763-201. A reference to the Southwest Research Institute report was added to C0-PP-001, Plant Partitioning Notebook.</p>	<p>the FPRA plant partitioning analysis.</p>
PP-B7-01	PP-B7 PP-C3 PP-C4 QLS-A1	Plant Partitioning Qualitative Screening	Complete	<p>1. The walk down nomenclature does not match the PP notebook. Example page 561 of the walkdown documentation uses nomenclature in the containment that does not match the PP notebook.</p> <p>2. There are many areas inaccessible such as: #23 Charging Pump Room, U1 Service Water Pump Room, U1 East Battery Room, E/W</p>	<p>A table was created to correlate the building or area nomenclature that was used for the plant walkdown documentation, to the plant analysis unit identifiers used in the Fire PRA analysis. This table was added to C0-PP-001, Calvert Cliffs Fire PRA Plant Partitioning Notebook as Table 17.</p> <p>The facilities and rooms that were not originally walked-down were reviewed. Supplemental walkdowns were performed and supplemental walkdown datasheets</p>	<p>No impact to ILRT analysis, as this affects the FPRA plant partitioning documentation.</p>

Table 2 Fire PRA Peer Review – Facts and Observations – Findings

F&O ID	SR	Topic	Status	Finding	Disposition	Impact on ILRT Analysis
				<p>Corridor. These areas appear to be accessible with a little effort. In some of the areas screened out in QLS, the areas were inaccessible and did not have a confirmatory walkdown. Finding level assessed due to the incompleteness of the walkdown documentation.</p>	<p>were generated. For areas that were not accessible at the time of the supplemental walkdowns (for radiological safety reasons, personnel safety concerns, or access otherwise denied), The reason for inaccessibility was added to Table 17.</p>	
				<p>1. Prepare a table that correlates the PAUs from the PP notebook with the area nomenclature used in the walkdown documentation. 2. Complete the walkdowns, particularly for areas screened in the QLS task.</p>		
CS-B1-01	CS-B1 CS-C4	Fire PRA Cable Selection and Location	Complete	<p>Current Breaker coordination study still in progress. This study needs to be completed in order to receive a category II met for CS-B1.</p> <p>Complete the breaker coordination study.</p>	<p>The breaker coordination study has been completed. As described in ECP-13-000321, Form 12, Engineering Evaluation, all PRA common power supplies are assumed to meet - or will meet - the coordination requirements of NFPA 805, except as noted in C0-CS-001, Fire PRA Cable Selection Notebook. As described in the cable selection notebook, two 120VAC lighting panels are not validated as coordinated, and these panels are assumed to fail for all Fire PRA scenarios. Also, as described in the PRA notebook a breaker for 480V motor control center</p>	<p>No impact to ILRT analysis, as this affects the FPRA plant Cable Selection analysis and the item has been completed.</p>

Table 2 Fire PRA Peer Review – Facts and Observations – Findings

F&O ID	SR	Topic	Status	Finding	Disposition	Impact on ILRT Analysis
					MCC101BT has not been validated as coordinated. This breaker, 52-10150, is modeled so that a fire-induced electrical fault on the breaker's power cabling will fail MCC101BT. Finally, the notebook identifies that selected 120V power panels have coordination issues, but that these will be addressed by design changes and referenced in Attachment S – Modifications and Implementation Items.	
PRM-B3-01	PRM-B3 PRM-B4 PRM-B5	Fire PRA/Plant Response Model	Complete	<p>The FPRA model did not address events involving loss of both HVAC trains to the MCR, long term heatup of MCR and need for operator actions outside the MCR to compensate for the loss of electronic controls in the MCR, which was assumed as a CCDP of 1.0 for the plant. The basis for excluding this potential Core Damage sequence was addressed in questions to the Calvert Cliffs PRA team. This sequence is a new sequence outside the current FPRA model logic trees.</p> <p>Consider using a combination of MCR heatup calculations to define the time when operators would leave MCR and consider</p>	<p>Loss of Control Room HVAC can affect the operability and availability of equipment in the control room and cable spreading room. As described in Calvert PRA System Analysis Notebooks C0-SY-002, C0-SY-017, and C0-SY-030, loss of HVAC is modeled to have the effect of increasing the failure rate of 120VAC and 125VDC instruments and controls in the cable spreading room. For the control room, degradation of the 125VDC system is used as a conservative surrogate for control room I&C degradation.</p> <p>Loss of Control Room HVAC and subsequent temperature increases may adversely affect operator responses. The model reflects degradation of human actions by the degradation of the 125VDC system used for instruments and controls. Loss of Control Room HVAC is not expected to cause abandonment by</p>	No impact to the ILRT analysis, as the loss of MCR HVAC modeling has been implemented in the models used in the ILRT analysis

Table 2 Fire PRA Peer Review – Facts and Observations – Findings

F&O ID	SR	Topic	Status	Finding	Disposition	Impact on ILRT Analysis
				a recovery action for restoring cooling the MCR.	<p>operations staff of the control room due to high temperatures. On complete loss of HVAC with no mitigation, such as no use of emergency fans, calculation CA02725 shows a CR temperature of 123 deg F at 24-hours. While this is a challenging environment, this temperature is assessed as insufficient to solely drive a complete CR abandonment scenario. NUREG/CR-6738 describes operational experience where operators will continue to occupy the control room even under severe environments.</p> <p>Operations staff says that in consideration of high temperatures in the control room, that Operations would do what was needed to keep the cores safe and covered. The site safety director says that for a temperature of 123 deg F, the site would implement a mitigation strategy which would include stay-times, assessment of individuals for heat-related conditions, use of ice vests, and call-in of additional qualified operations staff to rotate into the control room.</p> <p>The above discussion was included in C0-SY-030, Control Room HVAC PRA System Notebook.</p>	

Table 2 Fire PRA Peer Review – Facts and Observations – Findings

F&O ID	SR	Topic	Status	Finding	Disposition	Impact on ILRT Analysis
FSS-A5-01	FSS-A5	Fire Scenario Selection and Analysis	Complete	<p>A range of ignition source / target set combinations has been represented for unscreened PAUs. These combinations are identified in relevant calculation sheets for unscreened PAUs. In some PAUs, sub-PAUs are defined and damage from a potential fire within the sub-PAU is addressed. However, it is not clear how or why damage would be limited to the specified sub-PAU because there are no physical barriers between specified sub-PAUs. The documentation is such that it cannot be determined if the selected fire scenarios provide reasonable assurance that the risk contribution of each unscreened PAU can be characterized. Another issue that influences the potential for fire propagation across sub-PAU boundaries is that the temperature measurement locations specified in the detailed FDS fire modeling evaluations do not generally coincide with locations where maximum temperature are</p>	<p>FDS modeling was used for fire scenario evaluations in the Cable Spreading Rooms and Switchgear Rooms. In both cases, thermocouple location was adjusted as identified in F&O FSS-D3-02. For the CSR, consequences were divided into scenarios based on mitigation potential. First, if the scenario was suppressed by the Halon system then the limit of damage was based on what was predicted by FDS in terms of temperature and energy. If it was unsuppressed it went to total room burn, which assumes failure of all targets in the room, regardless of the initial scenario boundary. For the Switchgear Room FDS analysis, the analysis was updated to add clarity to the analysis. A discussion of the application of sub-PAUs has been added to Addendum 1 to C0-FSS-004, Fire PRA Detailed Fire Modeling Notebook. Damage was not limited to specified sub-PAUs. Specific examples of the treatment of fire growth and the application of sub-PAUs have been provided.</p> <p>As described in C0-FSS-004, the sub-PAU analysis included spatial information from walkdown, along with engineering judgment, to determine if fire sources could fail additional components, cables, or other combustibles, potentially leading to more damage to surrounding equipment or cables. For scenarios that leveraged FDT</p>	<p>No impact to the ILRT analysis, as this affects the FPR model and the item is complete.</p>

Table 2 Fire PRA Peer Review – Facts and Observations – Findings

F&O ID	SR	Topic	Status	Finding	Disposition	Impact on ILRT Analysis
				<p>expected (e.g., within the fire plume).</p> <p>As a consequence, for some fire scenarios damage to targets is not predicted when it should be based on the specified damage criteria. Some scenarios are screened on the basis of temperature measurements that do not represent conditions at targets within the fire plume. (See F&O FSS-D3-02) This could have a significant impact on the potential for fire propagation across sub-PAU boundaries and needs to be discussed more thoroughly.</p>	<p>modeling, the issue related to whether the analysis had correctly addressed the impact of transients along the edge of a boundary interface for a sub-PAU. A comparable consideration was also related to secondary combustion and oil fires. Resolution involved selection of several representative PAUs for a sensitivity study that expanded the existing sub-PAUs and examined secondary ignition potential.</p>	
FSS-A5-01	FSS-A5	Fire Scenario Selection and Analysis	Complete	<p>There were indications that Calvert Cliffs had the tools and information in place to properly evaluate the propagation of fires across the sub-PAU boundaries given no physical barriers but there were no examples showing that this evaluation was performed or any explicit descriptions of how they were performed in general. The concern here is that without an explicit</p>	<p>The PAUs were considered representative of the work performed based on several criteria. The analysis indicated that the methods mentioned were indeed appropriate. Sub-PAU impacts did not change from the expanded assessment and that secondary ignition was bounded by the existing analysis and was appropriately addressed. The analysis was incorporated into the documentation for C0-FSS-004.</p>	<p>No impact to the ILRT analysis, as this affects the FPRA model and the item is complete.</p>

Table 2 Fire PRA Peer Review – Facts and Observations – Findings

F&O ID	SR	Topic	Status	Finding	Disposition	Impact on ILRT Analysis
				<p>description of the process for evaluating the spread of fires across sub-PAU boundaries with no physical barriers and detailed examples, there is the potential that in the future, new people updating the PRA may not know that they have to evaluate this.</p>		
				<p>Calvert Cliffs needs to describe their process for evaluating fire growth and propagation between sub-PAUs and as applicable, between PAUs. Specific examples of the sub-PAU fire growth need to be provided. If fire propagation from sub-PAU to sub-PAU was not treated, Calvert Cliffs needs to evaluate all sub-PAUs to determine if there is any potential for fire spread and then model the potential for spreading fires and for damage occurring across sub-PAU boundaries.</p>		
FSS-D2-01	FSS-D2	Fire Scenario Selection and Analysis	Complete	Where used, the FDS model was generally used with a level of grid resolution that was below the level of grid resolution documented in the	FDS modeling was used for fire scenario evaluations in the Cable Spreading Rooms and Switchgear Rooms.	No impact to the ILRT analysis, as this affects the FPRA model

Table 2 Fire PRA Peer Review – Facts and Observations – Findings

F&O ID	SR	Topic	Status	Finding	Disposition	Impact on ILRT Analysis
				<p>NUREG-1824 Verification and Validation study for the FDS model. A validation study was not conducted to support the use of this lower level of grid resolution. Grid resolution has a bearing on the results of FDS calculations. Grid resolutions outside the validation range in NUREG-1824 should be justified and validated.</p> <p>Increase the level of grid resolution in the FDS PAU Fire Evaluations (C0-FSS-004 R1) so that the grid resolution is within the validation range documented in NUREG-1824.</p>	<p>For the Cable Spreading Room FDS fire scenarios, a grid study was performed on the updated FDS model. The study recommended a grid size that was within the range in NUREG/CR-1824. That grid size was used for CSR FDS scenario evaluations. The study and results were incorporated into C0-FSS-004, Fire PRA Detailed Fire Modeling Notebook.</p> <p>The Unit 1 27' and 45' Switchgear Rooms were updated to increase the level of grid resolution to a value that is within the validation range documented in NUREG/CR-1824. Results calculated in the Unit 1 FDS models were applied to Unit 2. Results of the updated model are incorporated into C0-FSS-004 as Addendum 1.</p>	<p>and the item is complete.</p>
FSS-D3-01	FSS-D3 FSS-B2 FSS-D4	Fire Scenario Selection and Analysis	Complete	<p>This SR is not met because detailed FDS fire modeling evaluations of PAUs 302, 306, 311, 317, 407 and 430 assume that material surfaces are "inert." As noted on p. 44 of C0-FSS-004 R1, this assumption was made "... so that no objects in the PAU or the PAU structure (walls, floor, or ceiling) itself would absorb any heat from the various fire scenarios, producing a more</p>	<p>FDS modeling was used for fire scenario evaluations in the Cable Spreading Rooms and Switchgear Rooms.</p> <p>For the Cable Spreading Room FDS fire scenarios, the Unit 1 CSR was modified to include actual material properties and sensitivity analysis. Actual material properties were used in the updated U1CSR FDS model rather than the prior use of "inert" material conditions. Adiabatic conditions were used for any items with material properties that are unknown or of</p>	<p>No impact to the ILRT analysis, as this affects the FPRA model and the item is complete.</p>

Table 2 Fire PRA Peer Review – Facts and Observations – Findings

F&O ID	SR	Topic	Status	Finding	Disposition	Impact on ILRT Analysis
				<p>conservative or worst case result for all fire scenarios' impacts to the components and cables within the PAU model. As such, no detailed material properties were required to be defined in FDS for the scenarios to function correctly." However, specification of material surfaces as "inert" in FDS does not prevent heat absorption into material surfaces. On the contrary, this specification maintains material surfaces at ambient temperature in FDS, which tends to maximize heat absorption into these surfaces. To prevent heat absorption into material surfaces, they should have been specified as "adiabatic" rather than as "inert." The "inert" parameter in FDS maximizes heat transfer to surfaces rather than minimize it. This can result in lower calculated gas temperatures.</p> <p>Specify materials surfaces as "adiabatic" rather than as "inert" in FDS to prevent them from absorbing heat in order to achieve the stated goal of</p>	<p>a high uncertainty to bound the analysis and prevent heat transfer into those objects. The CSR FDS model was executed and the results compared to the baseline results. This study was then documented in FSS-004. The results were applied to Unit 2 CSR. This study was then documented in FSS-004, Fire PRA Detailed Fire Modeling Notebook.</p> <p>The Unit 1 27' and 45' Switchgear Rooms were updated to specify representative material properties as referenced by NUREG 1805. This adjustment enabled the analysis to obtain more realistic estimates of environmental conditions for these fire scenarios. Results calculated in the Unit 1 FDS models were applied to Unit 2. Results of the updated model are incorporated into C0-FSS-004 as Addendum 1.</p>	

Table 2 Fire PRA Peer Review – Facts and Observations – Findings

F&O ID	SR	Topic	Status	Finding	Disposition	Impact on ILRT Analysis
				<p>producing a more conservative or worst case result. This may prove to be overly conservative, in which case specification of realistic material properties could be used to achieve more realistic estimates of environmental conditions for these fire scenarios.</p>		
FSS-D3-02	FSS-D3 FSS-A5	Fire Scenario Selection and Analysis	Complete	<p>Temperature measurement locations specified in the detailed FDS fire modeling evaluations do not generally coincide with locations where maximum temperature are expected (e.g., within the fire plume). As a consequence, for some fire scenarios damage to targets is not predicted when it should be based on the specified damage criteria. Some scenarios are screened on the basis of temperature measurements that do not represent conditions at targets within the fire plume.</p> <p>Re-run FDS simulations with temperature measurement probes located within the fire</p>	<p>FDS modeling was used for fire scenario evaluations in the Cable Spreading Rooms and Switchgear Rooms.</p> <p>For the Cable Spreading Room FDS fire scenarios, new measurement devices were included in the updated U1CSR FDS model. The thermocouples were placed directly above the fire source in the updated FDS model and the scenarios re-evaluated. The results were applied to Unit 2 CSR. This study and the results were then documented in FSS-004, Fire PRA Detailed Fire Modeling Notebook.</p> <p>The Unit 1 27' and 45' SWGR rooms were updated to alter the location of the thermocouples such that the centerline plume temperature was recorded and used to determine target impacts. Results</p>	<p>No impact to the ILRT analysis, as this affects the FPRA model and the item is complete.</p>

Table 2 Fire PRA Peer Review – Facts and Observations – Findings

F&O ID	SR	Topic	Status	Finding	Disposition	Impact on ILRT Analysis
				plume or use other fire modeling tools such as FDTs to calculate fire plume temperatures for these scenarios.	calculated in the Unit 1 FDS models were applied to Unit 2. Results of the updated model are incorporated into C0-FSS-004 as Addendum 1.	
FSS-D8-01	FSS-D8	Fire Scenario Selection and Analysis	Complete	Fire detection timing is evaluated for detailed fire modeling cases that use FDS. This fire detection timing is then used to estimate automatic fire suppression timing and fire brigade response timing for these scenarios. However, the fire detection timing is based on modeling that does not include obstructions located beneath the ceiling that could have an impact on fire detector response. The fire detection timing is also based on an unjustified assumption regarding the type of smoke detectors installed in the affected PAUs. Obstructions to the flow of fire gases can have an impact on smoke concentrations and velocities, which in turn influence smoke detector response. Without including such obstructions in fire modeling simulations, their	FDS modeling was used for fire scenario evaluations in the Cable Spreading Rooms and Switchgear Rooms. For the updated Cable Spreading Room FDS fire scenarios, cable tray obstructions were placed in the ceiling area of the updated U1CSR FDS model. Additional thermocouple and heat flux data recording devices were added to the U1CSR model under the new cable tray obstructions in the vicinity of the fire source. The scenarios were re-evaluated. The results were applied to Unit 2. A sensitivity study was also performed. The study and new scenario results were incorporated into C0-FSS-004, Fire PRA Detailed Fire Modeling Notebook. The Unit 1 27' and 45' SWGR rooms were also updated to include significant obstructions such as cable trays and beam pockets within the switchgear rooms. Results calculated in the Unit 1 FDS models were applied to Unit 2. Results and details of this analysis are documented in C0-FSS-004 as Addendum 1.	No impact to the ILRT analysis, as this affects the FPRA model and the item is complete.

Table 2 Fire PRA Peer Review – Facts and Observations – Findings

F&O ID	SR	Topic	Status	Finding	Disposition	Impact on ILRT Analysis
				<p>impact on fire detection times is not evaluated.</p>		
				<p>Include obstructions located beneath the ceiling for the affected fire scenarios in order to evaluate their impact on fire detection timing. Provide justification for the selection of the type of smoke detector specified in the FDS simulations for these fire scenarios.</p>		
FSS-F3-01	FSS-F3	Fire Scenario Selection and Analysis	Complete	<p>To achieve CC II/III for this SR, a quantitative assessment of the risk of the selected fire scenarios involving a) exposed structural steel and b) the presence of a high-hazard fire sources must be completed consistent with the FQ requirements including the collapse of the exposed structural steel and any attendant damage. Such an assessment has not been done or was not documented in a readily discernible manner. This has a potential impact on fire risk quantification.</p>	<p>The Turbine Building was reviewed for potential fire scenarios where structural steel can be adversely affected. From the scenarios examined, those that can damage structural steel were selected for further analysis. The frequency, severity factor and non-suppression probability of each scenario were developed and included in the Structural Failure Analysis Notebook. These impacts were then added to FRANX database and quantified as part of the final Fire PRA risk quantification in Fire Quantification Notebooks C0-FRQ-001 and C0-FRQ-002.</p>	<p>No impact to the ILRT analysis, as this affects the FPRA model and the item is complete.</p>

Table 2 Fire PRA Peer Review – Facts and Observations – Findings

F&O ID	SR	Topic	Status	Finding	Disposition	Impact on ILRT Analysis
				Complete a quantitative assessment of the risk of the selected exposed structural steel fire scenarios consistent with the FQ requirements.		
FSS-G4-01	FSS-G4	Fire Scenario Selection and Analysis	Complete	<p>An assessment of the effectiveness, reliability and availability of credited passive fire barrier features has not been documented in the multi-compartment analysis. To achieve a CC II capability assessment, the effectiveness, reliability and availability of credited passive fire barrier features must be assessed.</p> <p>Assess the effectiveness, reliability and availability of credited passive fire barrier features and document this assessment.</p>	<p>Generic probabilities were used for credited passive fire barrier features in the multi-compartment analysis. At Calvert Cliffs, the fire barriers are verified to be effective through test procedures. An unreliability value was applied to all normally closed doors that represents the probability of the door being propped open given a fire in the exposing compartment. The probability of finding a failed sealed wall penetration is assumed to be very small to warrant propagation scenarios. A discussion of the effectiveness, reliability, and availability of fire barriers was added to C0-FSS-008, Calvert Fire PRA Multi-Compartment Analysis.</p>	<p>No impact to the ILRT analysis, as this affects the FPR model and the item is complete.</p>
FSS-G5-01	FSS-G5	Fire Scenario Selection and Analysis	Complete	<p>The effectiveness, reliability and availability of credited active fire barrier features have not been quantified in the multi-compartment analysis. To achieve a CC II capability assessment, the effectiveness, reliability and availability of credited active fire barrier</p>	<p>Active fire barriers were evaluated as effective in studies used to support Appendix R analysis. An unreliability value has been applied to all normally open, self closing dampers and doors; A discussion of the effectiveness of credited active fire barriers was added to C0-FSS-008, Calvert Fire PRA Multi-Compartment Analysis.</p>	<p>No impact to the ILRT analysis, as this affects the FPR model and the item is complete.</p>

Table 2 Fire PRA Peer Review – Facts and Observations – Findings

F&O ID	SR	Topic	Status	Finding	Disposition	Impact on ILRT Analysis
				features must be quantified. Quantify the effectiveness, reliability and availability of credited active fire barrier features and document this assessment.		
HRA-B2-01	HRA-B2	Human Reliability Analysis	Complete	Improve documentation of the adverse operator actions needed to address the impact of grounded or shorted electrical buses that might have an impact on other plant buses if not isolated and re energized in the areas identified. Very difficult to find the information within the HRA notebook alone, because the actions are modeled as inputs to FRANX. Provide new tables listing the actions considered or references to specific locations.	C0-HRA-001, Fire Human Reliability notebook, was updated to detail the adverse operator actions added to the model following the fire AOP review process. Table 3 was added to Section 2.2 detailing each basic event, set to true (1.0) used in the model to annotate the adverse operator actions in the model. These include actions to de-energize electrical busses to isolate them from potential shorts and grounds. Table 2 shows the HFEs added to the model as part of the AOP review, including actions to restore AC power to busses lost due to fire failure sequences.	No impact to the ILRT analysis, as this affects the FPR model documentation and the item is complete.
HRA-E1-01	HRA-E1	Human Reliability Analysis	Complete	Documentation for what was done was very good, however, the details for not selecting any spurious alarms is not clear. The documentation of the adverse actions put into the	C0-HRA-001, Fire Human Reliability Notebook, was updated detailing the Alarm Response Procedure review process. Table 12 was expanded to show the ARP review of alarm impact and operator interview notes for CR annunciators that	No impact to the ILRT analysis, as this affects the FPR model and

Table 2 Fire PRA Peer Review – Facts and Observations – Findings

F&O ID	SR	Topic	Status	Finding	Disposition	Impact on ILRT Analysis
				<p>model as "true" are not in the HRA report, actions identified in the cutset reviews are not clearly identified, rational for not using specific HFEs in the RCP trip actions, for identifying actions from procedures and the process for assigning uncertainty range for the combos. Doesn't permit verification of the rational for judgments made in deciding what is in and out of the Fire HRA. Also, from the calculation viewpoint the need to know the use of all manpower requirements during early time after fire initiator for dependency analysis.</p>	<p>could result in a manual reactor trip. No annunciators were identified that would cause the operator to terminate a systems or components operation based solely on the alarm itself, but several were identified that could potentially result in the operator tripping the Unit unnecessarily.</p>	<p>documentation and the item is complete.</p>
				<p>Enhance documentation of the specific issues needed to reproduce the assumptions and calculations used in the HRA.</p>	<p>C0-HRA-001 was also updated to detail the adverse operator actions added to the model following the fire AOP review process. Table 3 was added to Section 2.2 detailing each basic event, set to true (1.0) used in the model to annotate the adverse operator actions in the model. These include actions to de-energize electrical busses to isolate them from potential shorts and grounds. Table 2 shows the HFEs added to the model as part of the AOP review, including actions to restore AC power to busses lost due to fire failure sequences.</p>	
					<p>New HFEs added as part of the cutset review process are identified in Table 1 of C0-HRA-001, Fire Human Action Reliability notebook. These are annotated with "identified during the development of the PRM Notebook." The cutset reviews are described in C0-QNS-001, Fire PRA Quantitative Screening Notebook. A new dependency analysis was performed after the new HFEs were added to the model,</p>	

Table 2 Fire PRA Peer Review – Facts and Observations – Findings

F&O ID	SR	Topic	Status	Finding	Disposition	Impact on ILRT Analysis
					<p>ensuring new dependency combinations are considered.</p> <p>Additional information was added to Table 1 of the Human Reliability Analysis Notebook, C0-HRA-001, detailing why each HFE was either retained or removed. For example, event FGAFW0SGTRISOL, Operator Feeds Affected SG with SGTR to Assure Heat Removal, was "Not retained for fire scenarios, because these actions are SGTR specific. Modeling was not necessary to ensure these actions did not appear in the cutsets, because the SGTR initiator is not being used for fire scenarios."</p> <p>Combination event multipliers are used in cutsets of multiple HEP actions to account for dependencies between HEP actions. To account for the uncertainty in HEP actions, an uncertainty parameter is added to the HEP action. When performing uncertainty analysis, the uncertainty parameters for combination events is increased proportionally when they are multiplied by the combination event multipliers.</p> <p>Based on interviews, there are sufficient non-control room personnel for fire recovery actions. Appendix D of C0-HRA-001 notes that there are no control room</p>	

Table 2 Fire PRA Peer Review – Facts and Observations – Findings

F&O ID	SR	Topic	Status	Finding	Disposition	Impact on ILRT Analysis
					<p>operators assigned to the fire brigade. There were no identified staffing issues or interferences between operators performing fire recovery actions and members of the fire brigade.</p>	
FQ-A1-01	FQ-A1	Fire Risk Quantification	Complete	<p>Treatment of 0 CCDPs scenarios is not clear and appears to result in an underestimate of total risk (the underestimate appears to be small based on the sensitivity evaluations performed):</p> <ul style="list-style-type: none"> 1 - with respect to opposite unit quantification, use CCDP for reactor trip initiator unless confirmation of no trip is documented; 2 - address use of 0 CCDP for control room HVAC loss scenarios, apply CCDP consistent with control room abandonment 3 - for scenarios with limited impact with a 0 CCDP, due to cutsets below truncation limit, apply a baseline CCDP based on reactor trip initiator <p>More than 50% of the scenarios have a 0 CCDP but no clear discussion of the</p>	<p>The fire risk quantification process has been updated in notebooks C0-FRQ-001 and C0-FRQ-002 to address the issue with FRANX fire scenarios having a zero conditional probability for CDF and LERF.</p> <ul style="list-style-type: none"> 1. When documented analysis shows that selected fire scenarios for one unit are screened from impact for the opposite unit (typically, no trip would be initiated), then that scenario may be excluded from the opposite unit's fire risk quantification. Otherwise, a nominal conditional probability, as described in item 3 below, would apply. 2. F&O PRM-B3-01 identifies the concern with loss of Control Room HVAC with control room abandonment. As discussed in more detail with the resolution to PRM-B3-01, subsequent investigation revealed that loss of CR HVAC is not expected to cause abandonment by the operations staff of the control room due to high temperatures. Loss of CR HVAC and subsequent temperature increases may adversely affect operator responses, and 	<p>No impact to the ILRT analysis, as this affects the FPR model and the item is complete.</p>

Table 2 Fire PRA Peer Review – Facts and Observations – Findings

F&O ID	SR	Topic	Status	Finding	Disposition	Impact on ILRT Analysis
				<p>basis for the 0 CCDP is provided.</p> <p>Treatment of 0 CCDPs scenarios:</p> <p>1 - with respect to opposite unit quantification, use CCDP for reactor trip initiator unless confirmation of no trip is documented;</p> <p>2 - address use of 0 CCDP for control room HVAC loss scenarios, apply CCDP consistent with control room abandonment</p> <p>3 - for scenarios with limited impact with a 0 CCDP, due to cutsets below truncation limit, apply a baseline CCDP based on reactor trip initiator</p>	<p>the model reflects degradation of human actions with loss of CR HVAC. C0-SY-030, Control Room HVAC PRA System Notebook, was updated to include this discussion.</p> <p>3. The new quantification process described in the FRQ notebooks is to assure a nominal conditional value is calculated for these low significant scenarios by 1) recalculating the zero-conditional scenarios at a lower truncation value to assure resolution in the scenario cutset file and conditional probabilities , and/or to 2) use a baseline conditional probability for CDF and LERF for the internal events reactor trip initiating vent - IE0PT for Unit 1 or IE0PT-2 for Unit 2</p>	
FQ-B1-01	FQ-B1	Fire Risk Quantification	Complete	<p>We observed zero CCDPs for some PAU CDF and LERF values in the FRANX tables (e.g., PAU 512) which eliminated loss of HVAC to the MCR as a potential MCR abandonment sequence.</p> <p>Treatment of 0 CCDPs scenarios:</p> <p>1 - with respect to opposite unit quantification, use CCDP for</p>	<p>The fire risk quantification process has been updated in notebooks C0-FRQ-001 and C0-FRQ-002 to address the issue with FRANX fire scenarios having a zero conditional probability for CDF and LERF.</p> <p>1. When documented analysis shows that selected fire scenarios for one unit are screened from impact for the opposite unit (typically, no trip would be initiated), then that scenario may be excluded from the</p>	<p>No impact to the ILRT analysis, as this affects the FPRA model and the item is complete.</p>

Table 2 Fire PRA Peer Review – Facts and Observations – Findings

F&O ID	SR	Topic	Status	Finding	Disposition	Impact on ILRT Analysis
				<p>reactor trip initiator unless confirmation of no trip is documented;</p> <p>2 - address use of 0 CCDP for control room HVAC loss scenarios, apply CCDP consistent with control room abandonment (F&O FQ-A1-01 (F))</p> <p>3 - for scenarios with limited impact with a 0 CCDP, due to cutsets below truncation limit, apply a baseline CCDP based on reactor trip initiator</p> <p>Allowing zero CCDPs allows scenarios in the fire model to quantify with no contribution to the CDF or LERF value and this under represents those frequencies especially when considering delta risk evaluations.</p> <p>Replace the zero entries with the lowest CCPD for a plant trip with only random failures of the safety equipment as in the internal events model. We discussed this with the Calvert Cliffs PRA team and some of the zeros are due to fire areas in one unit potentially contributing to the CCDP of the</p>	<p>opposite unit's fire risk quantification. Otherwise, a nominal conditional probability, as described in item 3 below, would apply.</p> <p>2. F&O PRM-B3-01 identifies the concern with loss of Control Room HVAC with control room abandonment. As discussed in more detail with the resolution to PRM-B3-01, subsequent investigation revealed that loss of CR HVAC is not expected to cause abandonment by the operations staff of the control room due to high temperatures. Loss of CR HVAC and subsequent temperature increases may adversely affect operator responses, and the model reflects degradation of human actions with loss of CR HVAC. C0-SY-030, Control Room HVAC PRA System Notebook, was updated to include this discussion.</p> <p>3. The new quantification process described in the FRQ notebooks is to assure a nominal conditional value is calculated for these low significant scenarios by 1) recalculating the zero-conditional scenarios at a lower truncation value to assure resolution in the scenario cutset file and conditional probabilities, and/or to 2) use a baseline conditional probability for CDF and LERF for the</p>	

Table 2 Fire PRA Peer Review – Facts and Observations – Findings

F&O ID	SR	Topic	Status	Finding	Disposition	Impact on ILRT Analysis
				opposite unit. With the exception of these cases a method for handling the zeros needed to be developed and applied in the frequency quantifications.	internal events reactor trip initiating vent - IE0PT for Unit 1 or IE0PT-2 for Unit 2	

B. ATTACHMENT 2

This attachment contains the release timing plots from NC-94-020 [Reference 18] for MAAP cases HRIF, GIOY, and MRIF. The MAAP release groups FREL_1 through FREL_2 correlate to the release categories shown in the figure below, and the MAAP outputs for each are shown in the figures on the following pages.

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Dr. Mahmoud Massoud

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August 17, 1993

Revised Release Category Definitions

Release Category	Description	Representation from Default MAAP Scheme
1	Noble Gases	Noble Gases (Group 1)
2	Iodine	CsI (Group 2)
3	Cesium + Rubidium	CsI + CsOH (Mole Fraction of Group 2 + Group 6)
4	Tellurium + Antimony	TeO ₂ + Te ₂ + Sb (Mole Fraction of Group 3, Group 11, + Group 10)
5	Strontium	SrO ₂ (Group 4)
6	Ruthenium, Rhodium, Molybdenum, + Technetium	MoO ₂ (Group 5)
7	Yttrium, Lanthanum, Zirconium, Niobium, Praseodymium, Neodymium, Americium, + Curium	La ₂ O ₃ + Pr ₂ O ₃ + Nd ₂ O ₃ + Sm ₂ O ₃ + Y ₂ O ₃ (Group 8)
8	Cerium, Neptunium, + Plutonium	CeO ₂ + UO ₂ + NpO ₂ + PuO ₂ (Mole Fraction of Group 9 + Group 12)
9	Barium	BaO (Group 7)

Figure 1 – Release Category Definitions from NC-94-020 [Reference 18].

NC 94-02P 230

plot Oct. 7, 1993 10:58:40 AM (fission product; tape 35; page 1 of 5)

L2-HRIF, Loss of Main Feedwater

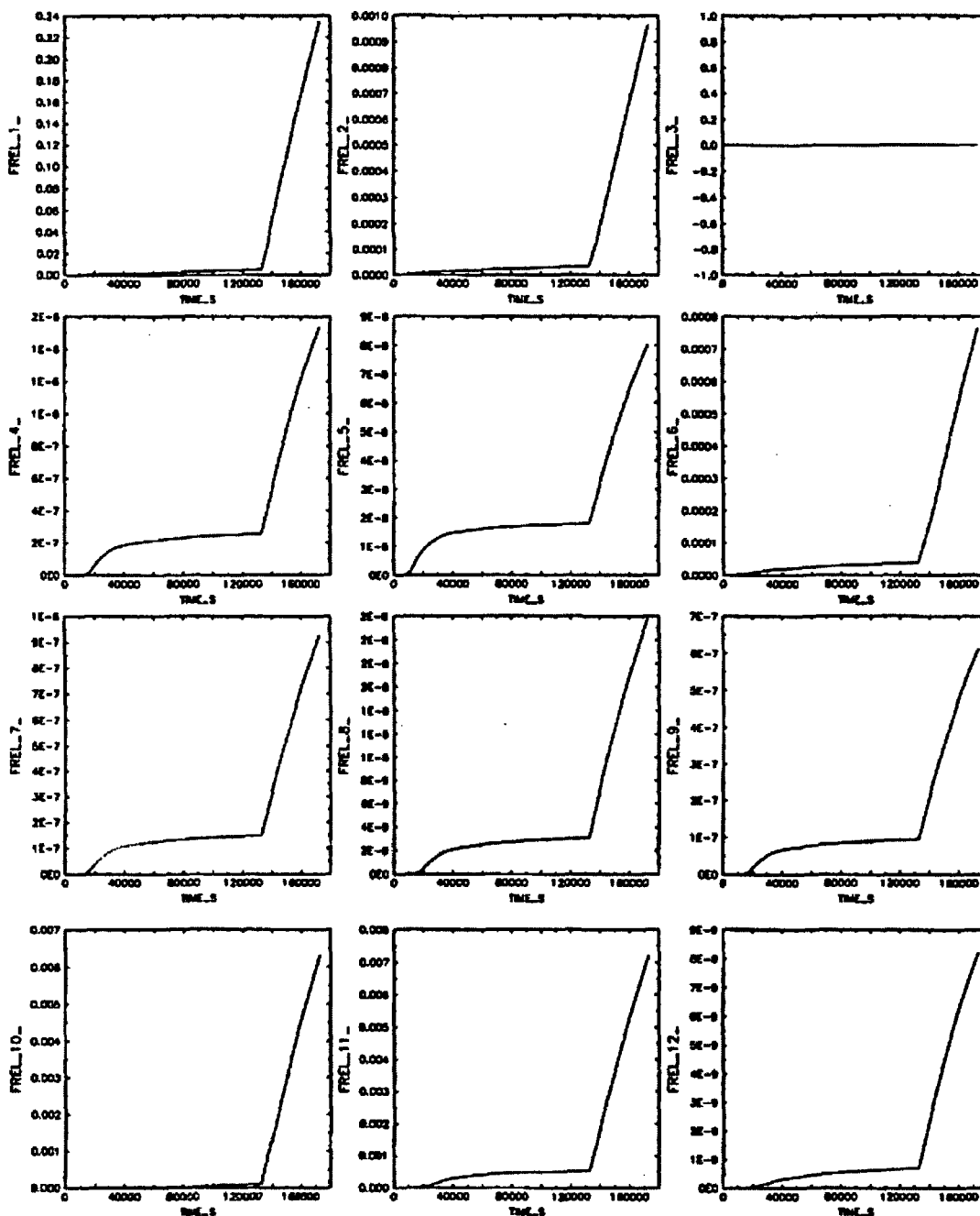


Figure 2 – MAAP Case HRIF Release Fractions

NC 94-020

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plot Dec. 5, 1993 5:39:04 PM

(fission product; tape 35; page 1 of 5)

L2-GIOY, Small LOCA

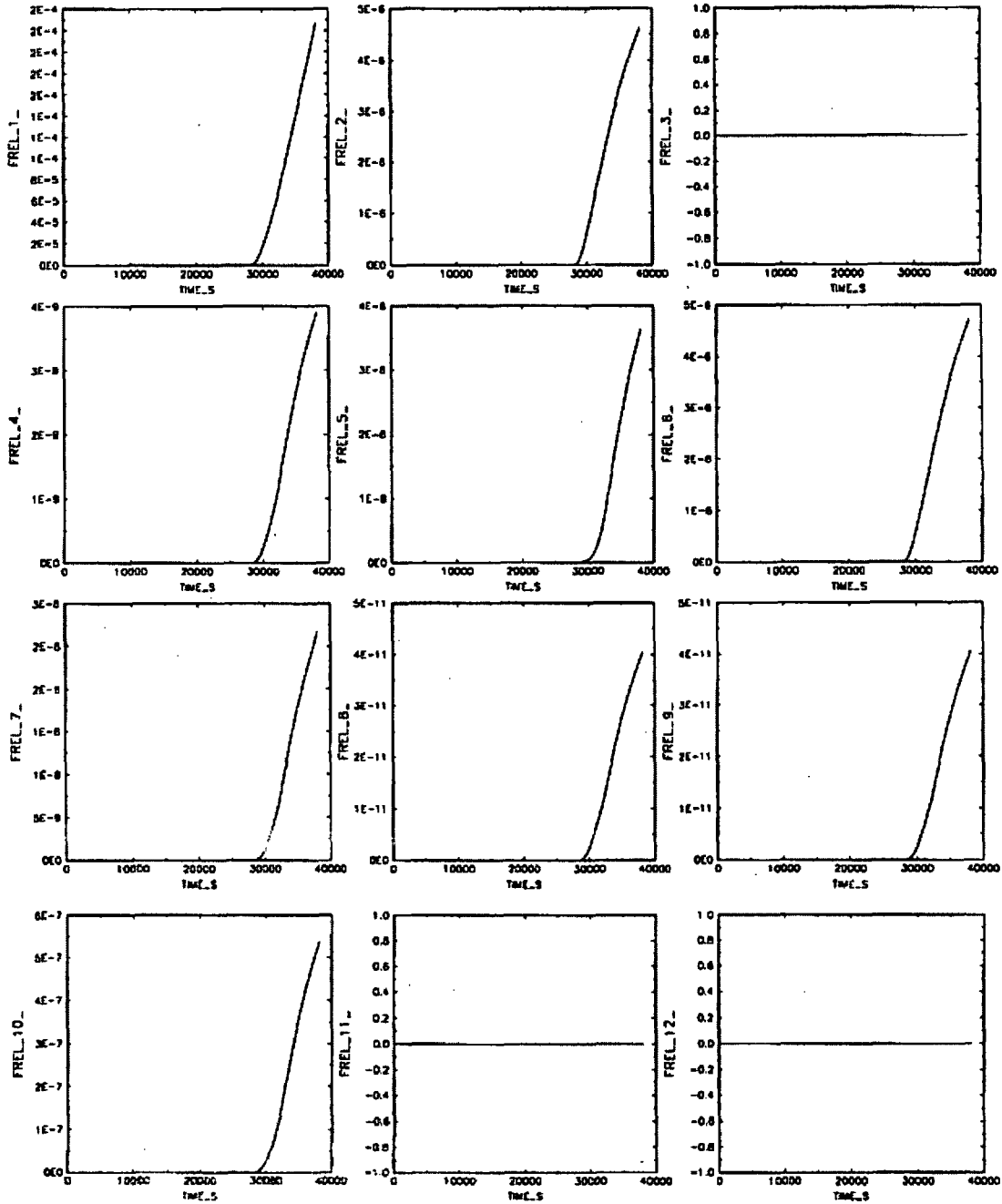


Figure 3 – MAAP Case GIOY Release Fractions

NC 94-020

337

plot Oct. 10, 1993 5:31:38 PM

(fission product; tape 35; page 1 of 5)
L2-MRIF, Small LOCA

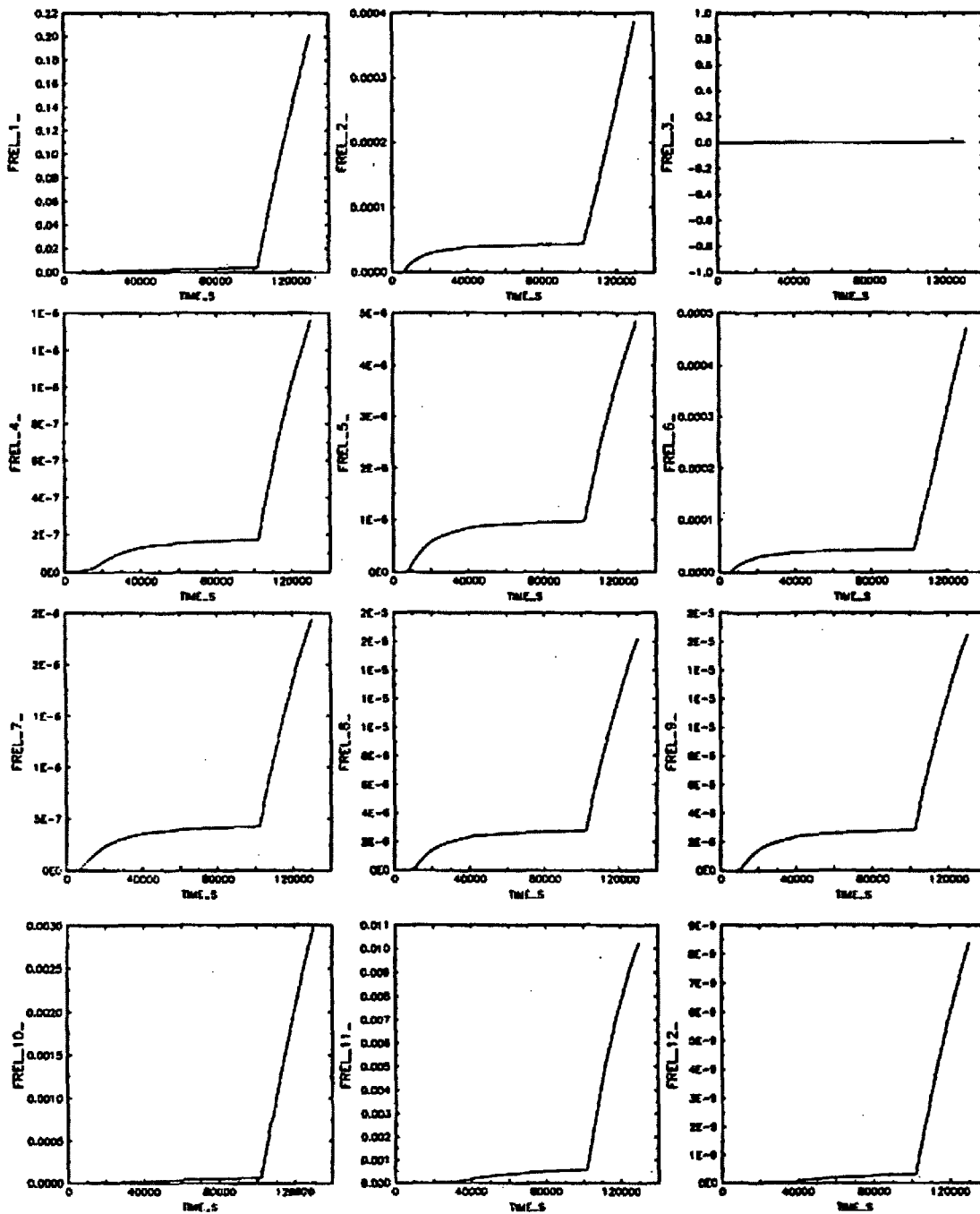


Figure 4 – MAAP Case MRIF Release Fractions

C. ATTACHMENT 3

Probabilistic Risk Assessment Licensing Branch (APLA) RAI 1

In the safety evaluation report for Electric Power Research Institute (EPRI) Technical Report (TR) 1009325, Revision 2, "Risk Impact Assessment of Extended Integrated Leak Rate Testing Intervals," the Nuclear Regulatory Commission (NRC) staff, in part, stated that for licensee requests for a permanent extension of the integrated leak rate testing (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."

Attachment 3 of the license amendment request (LAR) states that the 2010 full scope peer review of the internal events PRA model identified three supporting requirements (SRs) that were 'Not Met'. Table 1 of Attachment 3 to the LAR lists the findings from the 2010 peer review, but does not identify the three "Not Met" SRs. Identify which SRs were considered not met. For each SR, summarize why not meeting Capability Category I requirements will have no impact on the ILRT extension application.

Response

The 3 SRs that were noted as "not met" are LE-F2, LE-G5, and IFQU-A10. LE-F2 relates to LERF results. The dominant LERF contributors were reviewed and model changes implemented prior to the ILRT analysis. LE-G5 relates to the documentation of limitations of applications of the PRA. IFQU-A10 relates to documentation of the treatment of the internal flood analysis in the event trees. Therefore, none of the "not met" SRs impact the ILRT extension analysis.

APLA RAI 2

Section 5.3.2 of Attachment 3 to the LAR uses the Calvert Cliffs Nuclear Power Plant methodology from 2002 in evaluating the impact of steel liner corrosion on the extension of ILRT testing intervals. This assessment was based on two observed corrosion events at North Anna Power Station, Unit 2 and Brunswick Steam Electric Plant, Unit 2.

- a. If there have been additional instances of liner corrosion that could be relevant to this assessment, provide an updated list of observed corrosion events relevant to Calvert Cliffs containment, and an evaluation of the impact on risk results when all relevant corrosion events are included in the risk assessment.

Response

A search of the NRG website LER database identified two additional events have occurred since the Calvert Cliffs analysis was performed. In January 2000, a 3/16-inch circular through-liner hole was found at Cook Nuclear Plant Unit 2 caused by a wooden brush handle embedded immediately behind the containment liner. The other event occurred in April 2009, where a through-liner hole approximately 3/8-inch by 1-inch in size was identified in the Beaver Valley Power Station Unit 1 (BVPS-1) containment liner caused by pitting originating from the concrete side due to a piece of wood that was left behind during the original construction that came in contact with the steel liner. Two other containment liner through wall hole events occurred at Turkey Point Units 3 and 4 in October 2010 and November 2006, respectively. However, these events originated from the visible side caused by the failure of the

coating system, which was not designed for periodic immersion service, and are not considered to be applicable to this analysis. More recently, in October 2013, some through-wall containment liner holes were identified at BVPS-1, with a combined total area of approximately 0.395 square inches. The cause of these through wall liner holes was attributed to corrosion originating from the outside concrete surface due to the presence of rayon fiber foreign material that was left behind during the original construction and was contacting the steel liner. For risk evaluation purposes, these five total corrosion events occurring in 66 operating plants with steel containment liners over a 17.1 year period from September 1996 to October 4, 2013 (i.e., $5/(66 \cdot 17.1) = 4.43\text{E-}03$) are bounded by the estimated historical flaw probability based on the two events in the 5.5 year period of the Calvert Cliffs analysis (i.e., $2/(70 \cdot 5.5) = 5.19\text{E-}03$) incorporated in the EPRI guidance.

- b. Per EPRI TR-1009325, Revision 2, the risk metrics associated with the ILRT extension application include changes in large early release frequency (LERF), population dose, and conditional containment failure probability (CCFP). The steel liner corrosion assessment in Section 5.3.2 of Attachment 3 to the LAR calculates only the change in LERF. Include an estimate of change in population dose and CCFP due to increase in steel liner corrosion likelihood and demonstrate acceptability of the risk results.

Response

An estimate of change in population dose and CCFP due to increase in steel liner corrosion likelihood are provided in the following tables.

Unit 1 Steel Liner Corrosion Sensitivity CCFP						
	CCFP (3-per-10 year ILRT)	CCFP (1-per-10 year ILRT)	CCFP (1-per-15 year ILRT)	CCFP Increase (3-per-10 to 1-per-10)	CCFP Increase (3-per-10 to 1-per-15)	CCFP Increase (1-per-10 to 1-per-15)
Baseline CCFP	2.04E-01	2.09E-01	2.13E-01	5.11E-03	8.76E-03	3.65E-03
Corrosion Likelihood X 1000	2.04E-01	2.09E-01	2.13E-01	5.16E-03	8.84E-03	3.68E-03
Corrosion Likelihood X 10000	2.04E-01	2.10E-01	2.14E-01	5.57E-03	9.56E-03	3.98E-03
Corrosion Likelihood X 100000	2.06E-01	2.16E-01	2.23E-01	9.76E-03	1.67E-02	1.29E-02

Unit 2 Steel Liner Corrosion Sensitivity CCFP						
	CCFP (3-per-10 year ILRT)	CCFP (1-per-10 year ILRT)	CCFP (1-per-15 year ILRT)	CCFP Increase (3-per-10 to 1-per-10)	CCFP Increase (3-per-10 to 1-per-15)	CCFP Increase (1-per-10 to 1-per-15)
Baseline CCFP	2.01E-01	2.06E-01	2.09E-01	5.08E-03	8.71E-03	3.63E-03
Corrosion Likelihood X 1000	2.01E-01	2.06E-01	2.09E-01	5.13E-03	8.79E-03	3.66E-03
Corrosion Likelihood X 10000	2.01E-01	2.06E-01	2.10E-01	5.54E-03	9.50E-03	3.96E-03
Corrosion Likelihood X 100000	2.03E-01	2.12E-01	2.19E-01	9.70E-03	1.66E-02	6.93E-03

Unit 1 Steel Liner Corrosion Sensitivity Dose Rate						
	Dose Rate (3-per-10 year ILRT)	Dose Rate (1-per-10 year ILRT)	Dose Rate (1-per-15 year ILRT)	Dose Rate Increase (3-per-10 to 1-per-10)	Dose Rate Increase (3-per-10 to 1-per-15)	Dose Rate Increase (1-per-10 to 1-per-15)
Dose Rate	3.63E-02	1.21E-01	1.82E-01	8.48E-02	1.45E-01	6.06E-02
Corrosion Likelihood X 1000	3.67E-02	1.27E-01	2.04E-01	9.08E-02	1.67E-01	7.61E-02
Corrosion Likelihood X 10000	3.96E-02	1.84E-01	4.01E-01	1.45E-01	3.61E-01	2.17E-01
Corrosion Likelihood X 100000	6.94E-02	7.53E-01	2.37E+00	6.84E-01	2.30E+00	1.62E+00

Unit 2 Steel Liner Corrosion Sensitivity Dose Rate						
	CCFP (3-per-10 year ILRT)	CCFP (1-per-10 year ILRT)	CCFP (1-per-15 year ILRT)	CCFP Increase (3-per-10 to 1-per-10)	CCFP Increase (3-per-10 to 1-per-15)	CCFP Increase (1-per-10 to 1-per-15)
Dose Rate	1.96E-02	6.55E-02	9.82E-02	4.58E-02	7.86E-02	3.27E-02
Corrosion Likelihood X 1000	1.98E-02	6.89E-02	1.10E-01	4.91E-02	9.02E-02	4.12E-02
Corrosion Likelihood X 10000	2.14E-02	9.96E-02	2.17E-01	7.82E-02	1.95E-01	1.17E-01
Corrosion Likelihood X 100000	3.75E-02	4.07E-01	1.28E+00	3.70E-01	1.25E+00	8.76E-01

APLA RAI 3

Section 4.2.7 of 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.” EPRI TR-1009325, Revision 2-A further states that the “assessment can be taken from existing, previously submitted and approved analyses or another alternate method of assessing an order of magnitude estimate for contribution of the external event to the impact of the changed interval.” Section 5.3.1 in Attachment 3 to the LAR assesses the potential impact from external events contribution.

- a. The results of the seismic PRA performed for the Individual Plant Examinations for External Events (IPEEE) were used to assess the seismic risk with a reported core damage frequency (CDF) of $1.07E-5$ /year for both Unit 1 and Unit 2. Section 8, “Summary and Conclusions,” of the IPEEE report (Calculation No. RAN 97-031, IPEEE, Calvert Cliffs Nuclear Power Plant, “Individual Plant Examination of External Events,” August 1997) reports seismic CDF values of $1.29E-5$ /year for Unit 1 and $1.52E-5$ /year for Unit 2. Justify the use of the $1.07E-5$ /year seismic CDF value in the external events sensitivity study.

Response

Data from Table 3-6 of the IPEEE Seismic Analysis is used to calculate a Class 3b frequency due to seismic. As noted in Table 3-6 of the IPEEE Seismic Analysis, the values given in Table 3-6 reflect quantification without the surrogate top event LA. Top event LA represents seismic failure of rugged plant systems at a conservative screening fragility. Therefore, the total CDF is higher than the $1.07E-05$ /yr value given in Table 3-6. The CDF values given in Section 8.1 of the IPEEE Seismic Analysis are $1.29E-5$ /yr for Unit 1 and $1.52E-5$ /yr for Unit 2. The CDF contribution from surrogate top event LA was not included in the Unit 1 containment failure frequencies provided in the IPEEE (no containment failure frequencies are provided for Unit 2). In lieu of justification for the value of $1.07E-5$ /year seismic CDF, the contribution is conservatively added to the Unit 1 CDF of $1.07E-5$ /yr from Table 3-6 of the IPEEE to Containment Category I Failure (Intact). Note that the Intact category CDF is slightly rounded so that the total seismic CDF is preserved. Then, the percent each category contributes to the total CDF is calculated for the Unit 1 values and applied to the Unit 2 values because it is assumed that Unit 2 would have similar containment failure fractions to Unit 1. The resulting containment failure frequencies and total external events contribution using the revised IPEEE seismic CDF for each Unit are shown in the following tables:

Seismic Contribution to Frequencies of Containment Failure Categories			
Containment Failure Category	Unit 1 Seismic CDF (/yr)	Percent of CDF	Unit 2 Seismic CDF (/yr)
I. Intact Containment	2.69E-06	20.85%	3.17E-06
II. Late Containment Failure	8.63E-06	66.90%	1.02E-05
III. Early Small Containment Failure	1.70E-07 to 1.27E-06	1.32% to 9.84%	2.00E-07 to 1.50E-06
IV. Early Large Containment Failure	3.13E-07 to 1.41E-06	2.43% to 10.93%	3.69E-07 to 1.66E-06
V. Small Containment Bypass	0	0%	0
VI. Large Containment Bypass	0	0%	0
Total	1.29E-05		1.52E-05

CCNPP Unit 1 External Event Impact on ILRT LERF Calculation				
Hazard	EPRI Accident Class 3b Frequency			LERF Increase (from 3 per 10 years to 1 per 15 years)
	3 per 10 year	1 per 10 year	1 per 15 years	
External Events	5.13E-08	1.71E-07	2.57E-07	2.05E-07
Internal Events	1.14E-08	3.78E-08	5.68E-08	4.54E-08
Combined	6.27E-08	2.09E-07	3.13E-07	2.51E-07

CCNPP Unit 2 External Event Impact on ILRT LERF Calculation				
Hazard	EPRI Accident Class 3b Frequency			LERF Increase (from 3 per 10 years to 1 per 15 years)
	3 per 10 year	1 per 10 year	1 per 15 years	
External Events	6.52E-08	2.17E-07	3.26E-07	2.61E-07
Internal Events	6.14E-09	2.05E-08	3.07E-08	2.46E-08
Combined	7.13E-08	2.38E-07	3.57E-07	2.85E-07

- b. In Section 5.3.1.1 of Attachment 3 to the LAR the results from the IPEEE fire analysis were used to assess fire risk (CDF of $1.10E-5$ /year and LERF of $7.15E-7$ /year for Unit 2). Section 8, "Summary and Conclusions," of the IPEEE report (Calculation No. RAN 97-031, IPEEE, Calvert Cliffs Nuclear Power Plant, "Individual Plant Examination of External Events," August 1997) reports a Unit 2 fire CDF of $9.6E-5$ /year. Justify the use of selected IPEEE Unit 2 fire CDF/LERF values and discuss acceptability of Unit 2 risk results when using the IPEEE fire CDF/LERF.

Response

In lieu of justification of selected fire IPEEE CDF and LERF, the risk results have been revised using a Unit 2 fire CDF of $9.6E-5$ /year as given in Section 8 of the IPEEE. The Unit 2 containment failure frequencies and frequency 3b change are shown in the following tables:

Unit 2 Fire Contribution to Frequencies of Containment Failure Categories		
Containment Failure Category	Percentage	Unit 2 Fire CDF (/yr)
I. Intact Containment	36.4%	3.50E-05
II. Late Containment Failure	55.5%	5.33E-05
III. Early Small Containment Failure	1.7%	1.58E-06
IV. Early Large Containment Failure	6.5%	6.13E-06
V. Small Containment Bypass	0.0%	0.00E+00
VI. Large Containment Bypass	0.0%	0.00E+00
Total		9.60E-05

CCNPP Unit 2 External Event Impact on ILRT LERF Calculation				
Hazard	EPRI Accident Class 3b Frequency			LERF Increase (from 3 per 10 years to 1 per 15 years)
	3 per 10 year	1 per 10 year	1 per 15 years	
External Events	2.40E-07	8.00E-07	1.20E-06	9.61E-07
Internal Events	6.14E-09	2.05E-08	3.07E-08	2.46E-08
Combined	2.46E-07	8.21E-07	1.23E-06	9.88E-07

The internal event results are also provided to allow a composite value to be defined. When both the internal and external event contributions are combined the total change in Unit 1 and 2 LERF meet the guidance for small change in risk, as it exceeds the $1.0E-7/\text{yr}$ and remains less than $1.0E-6$ change in LERF for both units. For this change in LERF to be acceptable, total LERF must be less than $1.0E-5$.

Conservatively using the highest seismic LERF value and not crediting containment spray success or plant damage state adjustments for the Internal Events PRA, the total LERF values are calculated below:

$$\begin{aligned} \text{Unit 2: LERF}_{U2} &= \text{LERF}_{U2\text{internal}} + \text{LERF}_{U2\text{seismic}} + \text{LERF}_{U2\text{fire}} + \text{LERF}_{U2\text{class3Bincrease}} \\ &= 1.56E-6/\text{yr} + 1.66E-6/\text{yr} + 6.13E-6/\text{yr} + 9.84E-7/\text{yr} = 1.03E-5/\text{yr} \end{aligned}$$

The Unit 2 LERF is barely greater than $1.0E-5$. However, the Unit 2 Seismic LERF is between $3.69E-07$ and $1.66E-06$, and the highest Seismic LERF value was conservatively used to calculate $1.03E-5$. If the 74th percentile or smaller value of this range ($\leq 1.32E-6$) is used, the total Unit 2 LERF is less than $1.0E-5$. Moreover, the IPEEE does not include recent significant plant modifications designed specifically to reduce fire risk. Therefore, it is reasonable to conclude that the total Unit 2 LERF is less than $1.0E-5$. Since the total LERF for both units is less than $1.0E-5$, it is acceptable for the ΔLERF to be between $1.0E-7$ and $1.0E-6$.

APLA RAI 4

Section 5.2.4 of Attachment 3 to the LAR refines the calculation of the Class 3b frequencies for internal events by examining the source term. The conservatism in Class 3b frequency is reduced by analyzing the source term release time and defines an early release as occurring before 6.5 hours, which allows the removal of three accident scenarios from the Class 3b frequency. Elimination of these three scenarios appears to reduce the Class 3b frequency for internal events by a factor of 3 to 5. Section 5.3.1 of Attachment 3 to the LAR indicates that the same approach is used in the calculation of the fire Class 3b frequency in the external events sensitivity study when using the NFPA-805 fire PRA.

- a. Provide the calculated timing of the expected release for each of these three accident scenarios.

Response

The calculated timing of the expected release for the scenarios is as follows:

HRIF: 36 hours

GIOY: 8 hours

MRIF: 28 hours

- b. Provide the basis for the 6.5 hours delineation between early and late release.

Response

The 6.5 hour delineation between early and late release is based on the calculation of evacuation time estimates for Calvert Cliffs Nuclear Power Plant. Since early release timing is defined by time short enough that ability to evacuate nearby population is impaired such that a fatality is possible, and the calculation shows that

6.5 hours is sufficient for evacuation, this is chosen as the delineation between early and late release.

- c. Explain whether releases from these scenarios were included in the analysis to calculate the increase in the total integrated dose risk for all accident sequences.

Response

The releases from these scenarios were excluded in the analysis to calculate the total integrated dose risk, since the scenarios are excluded from the class 3b contribution based on either containment spray success or the late timing of the release.