Attachment 1

Evaluation of Risk Significance of Permanent ILRT Extension

(44 pages follow this cover sheet)

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Advancing the Science of Safety

Callaway Plant:
Evaluation of Risk Significance of Permanent ILRT Extension

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REVISION RECORD SUMMARY

Revision	Revision Summary
0	Initial Issue
1	Minor revision to wording in Section 5.2.9 to address client comment

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1.0 PURPOSE

The purpose of this analysis is to provide a risk assessment of permanently extending the currently allowed containment Type A Integrated Leak Rate Test (ILRT) from ten years to fifteen years. The extension would allow for substantial cost savings as the ILRT could be deferred for additional scheduled refueling outages for the Callaway Plant. The risk assessment follows the guidelines from NEI 94-01, Revision 3-A and 2-A [References 1 and 36], 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 Revision 3 as applied to ILRT interval extensions, 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], and the methodology used in EPRI 1018243, Revision 2-A of EPRI 1009325 [Reference 24].

2.0 SCOPE

Revisions to 10 CFR 50, 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 the limiting containment leakage rate of $1L_a$.

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 assessments 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" [Reference 2].

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), containment isolation failures contribute less than 0.1% 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 Callaway.

NEI 94-01 Revision 3-A supports using EPRI Report No. 1009325 Revision 2-A (EPRI 1018243), "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 [Reference 2]. 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 requires that visual examinations be conducted during at least three other outages, and in the outage during which the ILRT is being conducted. These requirements are not 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 [Reference 1].

In addition, the total annual risk (person-rem/yr 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 [Reference 6] and Safety Evaluations (SEs) for one-time interval extension (summarized in Appendix G of Reference 24) 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/yr and/or 0.002% to 0.46% of the total accident dose [Reference 24]. 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 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 [Reference 1].

3.0 REFERENCES

The following references were used in this calculation:

1. *Industry Guideline for Implementing Performance-Based Option of 10 CFR Part 50, Appendix J*, Revision 3-A, 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, Revision 3, January 2018.
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. PRA-IE-QUANT, At-Power Internal Events PRA, Quantification Analysis Notebook, Revision 3.
18. Calculation 17671-013, Callaway NFPA 805 Fire PRA, Integrated Fire Risk Report, Revision 2.

19. Callaway Plant Unit 1, Environmental Report for License Renewal, Appendix F – Severe Accident Mitigation Alternatives, 2010.
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. AMN#PES00042-REPT-002, "Callaway Energy Center Fire Probabilistic Risk Assessment Peer Review F&Os Closure Review," February 2021.
27. Enclosure 4, “License Amendment Request: Revise Technical Specifications to Adopt Risk-Informed Completion Times TSTF-505, Revision 2, ‘Provide Risk-Informed Extended Completion Times – RITSTF Initiative 4b’,” Information Supporting Justification of Excluding Sources of Risk Not Addressed by the PRA Models.
28. PRA-SEISMIC-QUANT, Seismic Probabilistic Risk Assessment, Quantification Analysis Notebook, Revision 2.
29. AMN#PES00031-REPT-003, "Callaway Energy Center Probabilistic Risk Assessment Focused Scope Peer Review," November 2020.
30. CALILRT.14-R141231B, “Integrated Leakage Rate Test Report,” October 12-14, 2014.
31. Surveillance Task Sheet S487844, “Containment Integrated Leak Rate Test,” September 28, 1999.
32. PRA-OEH-ANALYSIS, “Other External Hazards: Screening Assessment Notebook,” Revision 0.
33. NEI Letter to NRC, "Final Revision of Appendix X to NEI 05-04/07-12/12-16, Close-Out of Facts and Observations (F&Os)," (ADAMS Accession No. ML17086A431), dated February 21, 2017.
34. Technical Letter Report ML112070867, Containment Liner Corrosion Operating Experience Summary, Revision 1, August 2011.
35. Nuclear Regulatory Commission (NRC) Letter to Mr. Greg Krueger (NEI), "U.S. Nuclear Regulatory Commission Acceptance on Nuclear Energy Institute Appendix X to Guidance 05-04, 7-12, and 12-13, Close Out of Facts and Observations (F&Os)," May 3, 2017, Accession Number ML17079A427.
36. *Industry Guideline for Implementing Performance-Based Option of 10 CFR Part 50, Appendix J*, Revision 2-A, NEI 94-01, November 2008.

37. Regulatory Guide 1.200, “An Approach for Determining the Technical Adequacy of Probabilistic Risk Assessment Results for Risk-Informed Activities,” Revision 2, March 2009.
38. USNRC Memorandum, "US Nuclear Regulatory Commission Staff Expectations for an Industry Facts and Observations Independent Assessment Process," May 1st, 2017 (ADAMS Access ML17121A271).
39. USNRC Memorandum, “United States Nuclear Regulatory Commission Audit Report on Observation of Industry Independent Assessment Team Close-Out of Facts and Observations (F&Os),” May 1st, 2017 (ADAMS Access ML17095A252).
40. ULNRC-06690, "Enclosure 5: LAR Supplement to Address Audit Discussion Points Summarized in NRC Letter Dated September 14, 2021 (ML21238A138)."
41. PRA-FLOOD-QUANT, At-Power Internal Flooding PRA, Modeling and Quantification Analysis Notebook, Revision 2.
42. PRA-HW-QUANT, Quantification and Results of Plant Response Model, Revision 2.
43. RG 1.200, "Acceptability of Probabilistic Risk Assessment Results for Risk-Informed Activities," Revision 3, December 2020.
44. PWROG-19020-NP Revision 1, "Newly Developed Method Peer Review Pilot – General Screening Criteria for Loss of Room Cooling in PRA Modeling Risk Management Committee," PA-RMSC-1647, Revision 1, April 2020.
45. Nuclear Energy Institute (NEI) Topical Report (TR) NEI 06-09, "Risk-Informed Technical Specifications Initiative 4b, Risk-Managed Technical Specifications (RMTS) Guidelines," Revision 0-A, October 12, 2012 (ADAMS Accession No. ML 12286A322).
46. Letter from Jennifer M. Golder (NRC) to Biff Bradley (NEI), "Final Safety Evaluation for Nuclear Energy Institute (NEI) Topical Report (TR) NEI 06-09, "Risk-Informed Technical Specifications Initiative 4b, Risk-Managed Technical Specifications (RMTS) Guidelines," May 17, 2007 (ML071200238).
47. NEI 17-07, Revision 2, "Performance of PRA Peer Reviews Using the ASME/ANS PRA Standard," July 2019 (ADAMS Accession No. ML19228A242).
48. ASME/ANS RA-S-2009, Addenda to ASME/ANS RA-S-2008, "Standard for Level 1/Large Early Release Frequency Probabilistic Risk Assessment for Nuclear Power Plant Applications," February 2009.
49. PWROG-18027-NP Revision 0, "Loss of Room Cooling in PRA Modeling," April 2020.
50. PWROG-19012-P, "Peer Review of the Callaway Internal Events and Internal Flood Probabilistic Risk Assessment Model," April 2019.
51. PWROG-19034-P, "Independent Assessment of Facts & Observations Closure and Focused Scope Peer Review of the Callaway Probabilistic Risk Assessments," November 2019.
52. AMN#PES00031-REPT-001, "Callaway Energy Center Probabilistic Risk Assessment Focused Scope Peer Review," July 2020.
53. AMN#PES00031-REPT-002, "Callaway Energy Center Probabilistic Risk Assessment Peer Review F&Os Closure," July 2020.
54. PWROG-19022-P, "Peer Review of the Callaway External Hazard Screening Assessment and High Winds Probabilistic Risk Assessment," April 2019.

55. ASME/ANS RA-S CASE 1, Case for ASME/ANS RA-Sb-2013, "Standard for Level 1/Large Early Release Frequency Probabilistic Risk Assessment for Nuclear Power Plant Applications," ASME and ANS, November 2017.
56. PWROG-18044-P, "Peer Review of the Callaway Seismic Probabilistic Risk Assessment," June 2018.
57. NRC Letter, U.S. Nuclear Regulatory Commission Acceptance of ASME/ANS RA-S Case 1, March 12, 2018 (ADAMS access ML18017A964 and ML18017A966).
58. PWROG-19011-P, "Independent Assessment of Facts & Observations Closure and Focused Scope Peer Review of the Callaway Seismic Probabilistic Risk Assessment," March 2019.
59. NUREG/CR-6850 (also EPRI 1011989), "Fire PRA Methodology for Nuclear Power Facilities," September 2005, with Supplement 1 (EPRI 1019259), September 2010.
60. LTR-RAM-II-10-019, "Fire PRA Peer Review Against the Fire PRA Standard SRs From Section 4 of the ASME/ANS Standard for Level 1/Large Early Release Frequency Probabilistic Risk Assessments for Nuclear Power Plant Applications for The Callaway Nuclear Plant Fire PRA," October 2009.
61. AMN#PES00021-REPT-001, "Callaway Energy Center Fire Probabilistic Risk Assessment Peer Review F&Os Closure," June 2019.
62. Final Safety Analysis Report, Section 6.3.2.2: Net Positive Suction Head, Revision OL-25.
63. Calculation EJ-29, “Residual Heat Removal Pump NPSH Margin During Recirculation,” Revision 2.
64. NUREG/CR-6762, GSI-191 Technical Assessment: Parametric Evaluations for Pressurized Water Reactor Recirculation Sump Performance, US NRC, August 2002.
65. Letter from T. Witt (Ameren Missouri) to NRC, Response to Request for Additional Information Regarding Request for License Amendment and Regulatory Exemptions for risk-Informed Approach to Address GSI-191 and Respond to Generic Letter 2004-02 (LDCN 19-0014) (EPID L-2021-LLA-0059 and EPID L-2021-LLE-0021). (ULNRC-06735)

4.0 ASSUMPTIONS AND LIMITATIONS

The following assumptions were used in the calculation:

- The acceptability (i.e., technical adequacy) of the Callaway PRA [Reference 17] is consistent with the requirements of Regulatory Guide 1.200, as detailed in Appendix A.
- The Callaway Level 1 and 2 internal events PRA models [Reference 17] provide representative results.
- It is appropriate to use the Callaway internal events PRA model to effectively describe the risk change attributable to the ILRT extension. An analysis is performed in Section 5.2.7 to show the effect of including external event models for the ILRT extension [References 17, 18, 28, 40, and 41].
- Accident classes describing radionuclide release end states are defined consistent with EPRI methodology [Reference 24].
- The representative containment leakage for Class 1 sequences is $1L_a$. Class 3 accounts for increased leakage due to Type A inspection failures [Reference 24].
- The representative containment leakage for Class 3a sequences is $10L_a$ 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 $100L_a$ 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 in 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 [Reference 24].
- While precise numbers are maintained throughout the calculations, some values have been rounded when presented in this report. Therefore, summing individual values within tables may yield a different result than the sum result shown in the tables.

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 local leak rate test (LLRT) 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 Callaway. 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.

<u>NUREG/CR-3539 [Reference 10]</u>

Oak Ridge National Laboratory (ORNL) 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.

<u>NUREG/CR-4220 [Reference 11]</u>

NUREG/CR-4220 is a study performed by Pacific Northwest Laboratories for the NRC in 1985. The study reviewed over two thousand license event reports (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 0.02 person-rem per year…"

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

NUREG-1150 and the technical basis, NUREG/CR-4551 [Reference 7], 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.

<u>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]</u>

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.

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

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 basemat, each with a steel liner.

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

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 Callaway 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 Callaway ILRT Extension Risk Assessment includes the following:

- CDF and LERF Model results [References 17, 18, 28, 41, and 42]
- Dose within a 50-mile radius [Reference 19]
- ILRT results to demonstrate adequacy of the administrative and hardware issues [Reference 30 and 31]

Callaway Model

The Internal Events (IE) and Internal Flood (IF) PRA Models that are used for Callaway are characteristic of the as-built plant. The current CDF and LERF model is a linked fault tree model [Reference 17]. The IE+IF CDF is 7.08E-6/yr; the LERF is 3.20E-8/yr [References 17 and 41]. Table 5-1 and Table 5-2 provide a summary of the IE+IF CDF and LERF results for the Callaway PRA Model. Note: for the rest of this report, internal events risk includes internal floods.

Refer to Section 5.2.7 for further details on external events as they pertain to this analysis.

Table 5-1 – Internal Events CDF

Internal Events	Frequency (per year)
Internal Floods	5.11E-6
Transients	8.81E-7
FLB/SLB	7.45E-7
LOCAs	1.33E-7
LOOP	1.80E-7
SGTR	1.93E-8
ISLOCA	2.19E-9
Total Internal Events CDF	**7.08E-6**

Table 5-2 – Internal Events LERF

Internal Events	Frequency (per year)
Internal Floods	8.09E-9
Transients	1.40E-9
FLB/SLB	9.14E-10
LOCAs	1.54E-10
LOOP	2.24E-10
SGTR*	1.90E-8
ISLOCA	2.21E-9
Total Internal Events LERF	**3.20E-8**

*Note: induced SGTR (thermal or pressure induced) frequency is removed from the original initiator frequency and collectively reported as SGTR in this table (with nominal SGTR frequency).

Release Category Definitions

Table 5-3 defines the accident classes used in the ILRT extension evaluation, which is consistent with the EPRI methodology [Reference 24]. 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.

Table 5-3 – EPRI Containment Failure Classification [Reference 24]

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 5-3, 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 2 / 217 = 0.0092). For Class 3b, the probability is based on the Jeffreys non-informative prior for no “large” failures in 217 tests (i.e., 0.5 / (217+1) = 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 ΔLERF. NEI describes ways to demonstrate that, using plant-specific calculations, the ΔLERF 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 Callaway, as detailed in Section 5.2, involves subtracting LERF risk from the CDF that is applied to Class 3b because this portion of LERF is unaffected by containment integrity. To be consistent, the same change is made to the Class 3a CDF, even though these events are not considered LERF.

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 years / 2), and the average time that a leak could exist without detection for a ten-year interval is 5 years (10 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 Indian Point 3 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 and C local leak rate tests that will still occur). Eliminating this possibility conservatively over-estimates the factor increases attributable to the ILRT extension.

5.2 Analysis

The application of the approach based on the guidance contained in EPRI 1009325 [Reference 24] 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 5-4.

The analysis performed examined Callaway-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 1009325, Class 1 sequences [Reference 24]).
- 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 1009325, Class 3 sequences [Reference 24]).
- Accident sequences involving containment bypassed (EPRI 1009325, Class 8 sequences [Reference 24]), large containment isolation failures (EPRI 1009325, Class 2 sequences [Reference 24]), and small containment isolation “failure-to-seal” events (EPRI 1009325, Class 4 and 5 sequences [Reference 24]) 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 5-4 – 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 (SGTR and 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 5-4.

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 1009325 [Reference 24]). 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 5-4 were developed for Callaway by first determining the frequencies for Classes 1, 2, 6, 7, and 8.

Table 5-5 presents the grouping of each release category in EPRI Classes based on the associated description. Table 5-6 provides a summary of the accident sequence frequencies that can lead to radionuclide release to the public and have been derived consistent with the NEI Interim Guidance [Reference 3] and the definitions of accident classes 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 5.2.6.

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 < leakage < $10L_a$), and Class 3b is defined as a large liner breach ($10L_a$ < 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 was 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. Therefore, these LERF contributions from CDF are removed. The frequency of a Class 3a failure is calculated by the following equation:

$$Freq_{class3a} = P_{class3a} * (CDF - LERF) = \frac{2}{217} * (7.08\text{E-6} - 3.20\text{E-8}) = 6.49\text{E-8}$$

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

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

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

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

$$Freq_{class3b} = P_{class3b} * (CDF - LERF) = \frac{.5}{218} * (7.08\text{E-6} - 3.20\text{E-8}) = 1.62\text{E-8}$$

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 SAMA [Reference 19] provides the most recent plant-specific risk profile; since the model [Reference 17] does not calculate Intact frequency, the SAMA Intact frequency is scaled using the CDF, calculated below:

$$Freq_{Intact} = Intact_{SAMA}/CDF_{SAMA} * CDF_{IE} = 8.08\text{E-6} / 1.66\text{E-5} * 7.08\text{E-6} = 3.45\text{E-6}$$

The Intact frequency for internal events is 3.45E-6. The EPRI Accident Class 1 frequency is then adjusted by subtracting the EPRI Class 3a and 3b frequency (to preserve total CDF), calculated below:

$Freq_{class1} = Freq_{Intact} - (Freq_{class3a} - Freq_{class3b})$ = 3.45E-6 – (6.49E-8 – 1.62E-8) = 3.37E-6

Class 2 Sequences. This group consists of accident progression bins with large containment isolation failures. The large isolation failure is in internal events cutsets that contribute 0.242 of LERF. Multiplying by the total LERF, the EPRI Accident Class 2 frequency is 7.72E-9, as shown in Table 5-5.

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 occurs. 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. All other failure modes are bounded by the Class 2 assumptions. This accident class is also not evaluated further.

Class 7 Sequences. This group consists of all core damage accident progression bins in which containment failure is induced by severe accident phenomena (e.g., overpressure). This frequency is calculated by subtracting the Class 1, 2, and 8 frequencies from the total CDF. For this analysis, the frequency is determined from the EPRI Accident Class 7 frequency listed in Table 5-5.

Class 8 Sequences. This group consists of all core damage accident progression bins in which containment is bypassed via SGTR or ISLOCA. The SGTR internal events (including induced SGTR) cutsets contribute 0.593 of LERF. The ISLOCA initiators are in internal events cutsets that contribute 0.069 of LERF. Thus, the total EPRI Accident Class 8 frequency is the summation of the SGTR and ISLOCA frequencies, 2.12E-8, as shown in Table 5-5 and Table 5-6.

Table 5-5 – Accident Class Frequencies (Core Damage)

EPRI Category	Unit 1 Frequency (/yr)
Class 1	3.45E-6
Class 2	7.72E-9
Class 7	3.59E-6
Class 8 (SGTR)	1.90E-8
Class 8 (ISLOCA)	2.21E-9
Total (CDF)	7.08E-6

Table 5-6 – Baseline Risk Profile

Class	Description	Frequency (/yr)
1	No containment failure	3.37E-6[2]
2	Large containment isolation failures	7.72E-09
3a	Small isolation failures (liner breach)	6.49E-08
3b	Large isolation failures (liner breach)	1.62E-08
4	Small isolation failures - failure to seal (type B)	ε[1]
5	Small isolation failures - failure to seal (type C)	ε[1]
6	Containment isolation failures (dependent failure, personnel errors)	ε[1]
7	Severe accident phenomena induced failure (early and late)	3.59E-06
8	Containment bypass	2.12E-08
	Total	7.08E-06

1. ε represents a probabilistically insignificant value or a Class that is unaffected by the Type A ILRT.
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 population dose was calculated using data in Table 3-14 in Attachment F (Severe Accident Mitigation Alternatives (SAMA) analysis) of the Environmental Report [Reference 19]; its dose and frequency data is presented in Table 5-7. Reference 19 INTACT Release Category corresponds to EPRI Accident Class 1. LERF-CI (Containment Isolation Failure) Release Category corresponds to EPRI Accident Class 2. Since they are not associated with other classes, three containment end-states correspond to EPRI Accident Class 7 (LATE-COP, LATE-BMT, and LERF-CF Release Categories); the EPRI Accident Class 7 dose is calculated via a weighted average using the frequencies provided in Reference 19. The LERF-SG (Steam Generator Tube Rupture) and LERF-ITR (Induced Steam Generator Tube Rupture) Release Categories and LERF-IS (ISLOCA) Release Category correspond to EPRI Accident Class 8; dose used in this analysis is weighted via the ISLOCA and SGTR frequencies in this calculation. Class 3a and 3b population dose values are calculated from the Class 1 population dose and are represented as $10L_a$ and $100L_a$, respectively, as guidance in Reference 1 dictates. Since population dose (person-rem) is not presented directly in SAMA Table F.3-3, population dose rate (person-rem/yr) is divided by frequency (/yr) to calculate population dose.

Table 5-7 – Baseline Population Doses

Release Category	INTACT	LATE-COP	LATE-BMT	LERF-SG	LERF-ITR	LERF-IS	LERF-CI	LERF-CF	Total
Frequency (/yr)	8.08E-06	3.19E-06	2.55E-06	2.33E-06	2.17E-07	1.73E-07	1.66E-10	1.13E-08	1.66E-05
Population Dose Rate (person-rem/yr)	2.31E-02	1.72E+00	9.92E-02	2.13E+00	2.67E-01	3.46E-01	1.27E-04	9.27E-03	4.60E+00
Population Dose (person-rem)	2.86E+03	5.41E+05	3.89E+04	9.13E+05	1.23E+06	2.00E+06	7.66E+05	8.24E+05	-
EPRI Class	1	7	7	8	8	8	2	7	

Table 5-8 presents dose exposures calculated from the methodology described in Reference 24. Table 5-9 presents the baseline risk profile for Callaway.

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] and the Class 1 dose as follows:

$$EPRI\ Class\ 3a\ Population\ Dose = 10 * 2.86E{+}3 = 2.86E{+}4$$

$$EPRI\ Class\ 3b\ Population\ Dose = 100 * 2.86E{+}3 = 2.86E{+}5$$

Table 5-8 – Baseline Population Doses

Class	Description	Population Dose (person-rem)
1	No containment failure	2.86E+03
2	Large containment isolation failures	7.66E+05
3a	Small isolation failures (liner breach)	2.86E+04[1]
3b	Large isolation failures (liner breach)	2.86E+05[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)	N/A
7	Severe accident phenomena induced failure (early and late)	3.19E+05
8	Containment bypass	1.09E+06

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

Table 5-9 – Baseline Risk Profile for ILRT

Class	Description	Frequency (/yr)	Contribution (%)	Population Dose (person-rem)	Population Dose Rate (person-rem/yr)
1	No containment failure[2]	3.37E-06	47.67%	2.86E+03	9.65E-03
2	Large containment isolation failures	7.72E-09	0.11%	7.66E+05	5.91E-03
3a	Small isolation failures (liner breach)	6.49E-08	0.92%	2.86E+04	1.86E-03
3b	Large isolation failures (liner breach)	1.62E-08	0.23%	2.86E+05	4.62E-03
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)	ε[1]	ε[1]	ε[1]	ε[1]
7	Severe accident phenomena induced failure (early and late)	3.59E-06	50.77%	3.19E+05	1.15E+00
8	Containment bypass	2.12E-08	0.30%	1.09E+06	2.31E-02
	Total	7.08E-06	100.00%		1.19E+00

1. ε represents a probabilistically insignificant value or a Class that is unaffected by the Type A ILRT.
2. The Class 1 frequency is reduced by the frequency of Class 3a and Class 3b to preserve total CDF.

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 a 3-year interval (i.e., a simplified representation of a 3-in-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:

$$Freq_{Class3a10yr} = \frac{10}{3} * \frac{2}{217} * (CDF - LERF) = \frac{10}{3} * \frac{2}{217} * 7.04\text{E-}6 = 2.16\text{E-}7$$

$$Freq_{Class3b10yr} = \frac{10}{3} * \frac{.5}{218} * (CDF - LERF) = \frac{10}{3} * \frac{.5}{218} * 7.04\text{E-}6 = 5.38\text{E-}8$$

The results of the calculation for a 10-year interval are presented in Table 5-10.

Table 5-10 – 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[2]	3.18E-06	45.00%	2.86E+03	9.11E-03
2	Large containment isolation failures	7.72E-09	0.11%	7.66E+05	5.91E-03
3a	Small isolation failures (liner breach)	2.16E-07	3.06%	2.86E+04	6.19E-03
3b	Large isolation failures (liner breach)	5.38E-08	0.76%	2.86E+05	1.54E-02
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)	ε[1]	ε[1]	ε[1]	ε[1]
7	Severe accident phenomena induced failure (early and late)	3.59E-06	50.77%	3.19E+05	1.15E+00
8	Containment bypass	2.12E-08	0.30%	1.09E+06	2.31E-02
	Total	7.08E-06			1.21E+00

1. ε represents a probabilistically insignificant value or a Class that is unaffected by the Type A ILRT.
2. The Class 1 frequency is reduced by the frequency of Class 3a and Class 3b to preserve total CDF.

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:

$$Freq_{Class3a15yr} = \frac{15}{3} * \frac{2}{217} * (CDF - LERF) = 5 * \frac{2}{217} * 7.04\text{E-}6 = 3.25\text{E-}7$$

$$Freq_{Class3b15yr} = \frac{15}{3} * \frac{.5}{218} * (CDF - LERF) = 5 * \frac{.5}{218} * 7.04\text{E-}6 = 8.08\text{E-}8$$

The results of the calculation for a 15-year interval are presented in Table 5-11.

Table 5-11 – 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[2]	3.05E-06	43.09%	2.86E+03	8.72E-03
2	Large containment isolation failures	7.72E-09	0.11%	7.66E+05	5.91E-03
3a	Small isolation failures (liner breach)	3.25E-07	4.59%	2.86E+04	9.28E-03

Table 5-11 – Risk Profile for Once in 15 Year ILRT

Class	Description	Frequency (/yr)	Contribution (%)	Population Dose (person-rem)	Population Dose Rate (person-rem/yr)
3b	Large isolation failures (liner breach)	8.08E-08	1.14%	2.86E+05	2.31E-02
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)	ε[1]	ε[1]	ε[1]	ε[1]
7	Severe accident phenomena induced failure (early and late)	3.59E-06	50.77%	3.19E+05	1.15E+00
8	Containment bypass	2.12E-08	0.30%	1.09E+06	2.31E-02
	Total	7.08E-06			1.22E+00

1. ε represents a probabilistically insignificant value or a Class that is unaffected by the Type A ILRT.
2. The Class 1 frequency is reduced by the frequency of Class 3a and Class 3b to preserve total CDF.

5.2.4 Step 4 – Determine the Change in Risk in Terms of Internal Events 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}/yr and increases in LERF less than 10^{-7}/yr, and “small” changes in LERF as less than 10^{-6}/yr. Since Callaway does not rely on containment overpressure for net positive suction head (NPSH) for ECCS injection [References 62 and 63], the ILRT extension does not impact CDF (see Section 5.2.9 for details). Therefore, the relevant risk-impact metric is LERF.

For Callaway, 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 Table 5-10, the Class 3b frequency is 5.38E-8/yr; based on a 15-year test interval from Table 5-11, the Class 3b frequency is 8.08E-8/yr. 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 6.46E-8/yr. Similarly, the increase due to increasing the interval from 10 to 15 years is 2.69E-8/yr. Table 5-12 summarizes these results.

Table 5-12 – Impact on LERF due to Extended Type A Testing Intervals

ILRT Inspection Interval	3 Years (baseline)	10 Years	15 Years
Class 3b (Type A LERF)	1.62E-08	5.38E-08	8.08E-08
ΔLERF (3 year baseline)		3.77E-08	6.46E-08
ΔLERF (10 year baseline)			2.69E-08

As can be seen, even with the conservatisms included in the evaluation (per the EPRI methodology), the estimated change in LERF meets the criteria for a "very small" change when comparing the 15-year results to the current 10-year requirement and when comparing the 15-year results to the original 3-year requirement, as it remains less than 1.0E-7/yr in both cases.

NEI 94-01 Revision 2-A [Reference 36] states that a "small" population dose is defined as an increase of ≤ 1.0 person-rem/yr, or ≤ 1% of the total population dose, whichever is less restrictive for the risk impact assessment of the extended ILRT intervals. As shown in Table 5-13, the results of this calculation meet the dose rate criteria.

Table 5-13 – Impact on Dose Rate due to Extended Type A Testing Intervals

ILRT Inspection Interval	10 Years	15 Years
ΔDose Rate (3 year baseline)	1.457E-02	2.498E-02
ΔDose Rate (10 year baseline)		1.041E-02
%ΔDose Rate (3 year baseline)	1.224%	2.098%
%ΔDose Rate (10 year baseline)		0.863%

1. ΔDose Rate is the difference in the total dose rate between cases. For instance, 'ΔDose Rate (3 year baseline)' for the 1 in 15 case is the total dose rate of the 1 in 15 case minus the total dose rate of the 3 in 10 year case.
2. %ΔDose Rate is the ΔDose Rate divided by the total baseline dose rate. For instance, '%ΔDose Rate (3 year baseline)' for the 1 in 15 case is the 'ΔDose Rate (3 year baseline)' of the 1 in 15 year case divided by the total dose rate of the 3 in 10 year case.

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

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. Table 5-14 shows the steps and results of this calculation.

Table 5-14 – Impact on CCFP due to Extended Type A Testing Intervals

ILRT Inspection Interval	3 Years (baseline)	10 Years	15 Years
f(ncf) (/yr)	3.44E-06	3.40E-06	3.37E-06
f(ncf)/CDF	0.486	0.481	0.477
CCFP	0.514	0.519	0.523
ΔCCFP (3 year baseline)		0.533%	0.913%
ΔCCFP (10 year baseline)			0.381%

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 0.913%. Therefore, this increase is judged to be "small."

5.2.6 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 a methodology similar to the Calvert Cliffs liner corrosion analysis [Reference 5]. The Calvert Cliffs analysis was performed for a concrete cylinder and dome and a concrete basemat, 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 basemat 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

<u>Assumptions</u>

- Consistent with the Calvert Cliffs analysis, a half failure is assumed for basemat concealed liner corrosion due to the lack of identified failures (See Table 5-15, Step 1).
- In the 5.5 years following September 1996 when 10 CFR 50.55a started requiring visual inspection, there were three events where a through wall hole in the containment liner was identified. These are Brunswick 2 on 4/27/99, North Anna 2 on 9/23/99, and D. C. Cook 2 in November 1999. The corrosion associated with the Brunswick event is believed to have started from the coated side of the containment liner. Although Callaway has a different containment type, this event could potentially occur at Callaway (i.e., corrosion starting on the coated side of containment). Construction material embedded in the concrete may have contributed to the corrosion. The corrosion at North Anna is believed to have started on the uninspectable side of containment due to wood imbedded in the concrete during construction. The D.C. Cook event is associated with an inadequate repair of a hole drilled through the liner during construction. Since the hole was created during construction and not caused by corrosion, this event does not apply to this analysis. Based on the above data, there are two corrosion events from the 5.5 years that apply to Callaway.
- 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) (See Table 5-4, 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 5-15, 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 basemat. These values were determined from an assessment of the probability versus containment pressure. For Callaway, the ILRT maximum pressure is 48.8 psig [Reference 30]. Probabilities of 1% for the cylinder and dome, and 0.1% for the basemat are used in this analysis, and sensitivity studies are included in Section 5.3.1 (See Table 5-15, Step 4).
- Consistent with the Calvert Cliffs analysis, the likelihood of leakage escape (due to crack formation) in the basemat region is considered to be less likely than the containment cylinder and dome region (See Table 5-15, Step 4).
- In the Calvert Cliffs analysis, it is noted that approximately 85% of the interior wall surface is accessible for visual inspections. 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 5-15, 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 5-15 – Steel Liner Corrosion Base Case

Step	Description	Containment Cylinder and Dome (85%)	Containment Basemat (15%)
1	Historical liner flaw likelihood Failure data: containment location specific Success data: based on 70 steel-lined containments and 5.5 years since the 10 CFR 50.55a requirements of periodic visual inspections of containment surfaces	Events: 2 (Brunswick 2 and North Anna 2) 2 / (70 x 5.5) = 5.19E-03	Events: 0 Assume a half failure 0.5 / (70 x 5.5) = 1.30E-03
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 — Failure rate 1 — 2.05E-03 average 5-10 — 5.19E-03 15 — 1.43E-02 15 year average = 6.44E-03	Year — Failure rate 1 — 5.13E-04 average 5-10 — 1.30E-03 15 — 3.57E-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.	0.71% (1 to 3 years) 4.14% (1 to 10 years) 9.66% (1 to 15 years)	0.18% (1 to 3 years) 1.04% (1 to 10 years) 2.42% (1 to 15 years)
4	Likelihood of breach in containment given liner flaw	1%	0.1%
5	Visual inspection detection failure likelihood	10% 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.	100% Cannot be visually inspected
6	Likelihood of non-detected containment leakage (Steps 3 x 4 x 5)	0.00071% (3 years) 0.71% x 1% x 10% 0.00414% (10 years) 4.14% x 1% x 10% 0.00966% (15 years) 9.66% x 1% x 10%	0.00018% (3 years) 0.18% x 0.1% x 100% 0.00104% (10 years) 1.04% x 0.1% x 100% 0.00242% (15 years) 2.42% x 0.1% x 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 basemat, as summarized below for Callaway.

Table 5-16 – Total Likelihood on Non-Detected Containment Leakage Due to Corrosion for Callaway

Description
At 3 years: 0.00071% + 0.00018% = 0.00089%
At 10 years: 0.00414% + 0.00104% = 0.00517%
At 15 years: 0.00966% + 0.00242% = 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 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 NRC 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 [Reference 34]. 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*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*5.5) = 5.19E-03) incorporated in the EPRI guidance [Reference 34].

5.2.7 Impact from External Events Contribution

An assessment of the impact of external events is performed. The primary purpose 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.

The Fire PRA calculated a CDF of 1.25E-5 and a LERF of 4.07E-8 [Reference 18]. As described in Section 5.1.3, additional consideration is made to not apply failure probabilities on those cases that are already LERF scenarios. Therefore, the LERF contributions are removed from CDF; to reduce conservatism in the ILRT extension analysis, the methodology of subtracting existing LERF from CDF is also applied to the Fire PRA model. The following shows the calculation for Class 3b:

$$Freq_{class3b} = P_{class3b} * (CDF - LERF) = \frac{0.5}{218} * (1.25E\text{-}5 - 4.07E\text{-}8) = 2.85\text{E-}8$$

$$Freq_{class3b10yr} = \frac{10}{3} * P_{class3b} * (CDF - LERF) = \frac{10}{3} * \frac{0.5}{218} * (1.25E\text{-}5 - 4.07E\text{-}8) = 9.51\text{E-}8$$

$$Freq_{class3b15yr} = \frac{15}{3} * P_{class3b} * (CDF - LERF) = 5 * \frac{0.5}{218} * (1.25E\text{-}5 - 4.07E\text{-}8) = 1.43\text{E-}7$$

The Seismic PRA calculated a CDF of 4.56E-5 and a LERF of 3.41E-6 using ACUBE [Reference 28]. As described in Section 5.1.3, additional consideration is made to not apply failure probabilities on those cases that are already LERF scenarios. Subtracting seismic LERF from CDF, the Class 3b frequency can be calculated by the following formulas:

$$Freq_{class3b} = P_{class3b} * (CDF - LERF) = \frac{0.5}{218} * (4.56E\text{-}5 - 3.41E\text{-}6) = 9.68\text{E-}8$$

$$Freq_{class3b10yr} = \frac{10}{3} * P_{class3b} * (CDF - LERF) = \frac{10}{3} * \frac{0.5}{218} * (4.56E\text{-}5 - 3.41E\text{-}6) = 3.23\text{E-}7$$

$$Freq_{class3b15yr} = \frac{15}{3} * P_{class3b} * (CDF - LERF) = 5 * \frac{0.5}{218} * (4.56E\text{-}5 - 3.41E\text{-}6) = 4.84\text{E-}7$$

The High Winds (HW) PRA calculated a CDF of 2.79E-6 and a LERF of 7.98E-9 using ACUBE [Reference 42]. As described in Section 5.1.3, additional consideration is made to not apply failure probabilities on those cases that are already LERF scenarios. Subtracting seismic LERF from CDF, the Class 3b frequency can be calculated by the following formulas:

$$Freq_{class3b} = P_{class3b} * (CDF - LERF) = \frac{0.5}{218} * (2.79E\text{-}6 - 7.98E\text{-}9) = 6.39\text{E-}9$$

$$Freq_{class3b10yr} = \frac{10}{3} * P_{class3b} * (CDF - LERF) = \frac{10}{3} * \frac{0.5}{218} * (2.79E\text{-}6 - 7.98E\text{-}9) = 2.13\text{E-}8$$

$$Freq_{class3b15yr} = \frac{15}{3} * P_{class3b} * (CDF - LERF) = 5 * \frac{0.5}{218} * (2.79E\text{-}6 - 7.98E\text{-}9) = 3.19\text{E-}8$$

The external event 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 Table 5-17.

Table 5-17 – Callaway 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)	LERF Increase (from 1 per 10 years to 1 per 15 years)
	3 per 10 years	1 per 10 years	1 per 15 years		
External Events	1.32E-07	4.39E-07	6.59E-07	5.27E-07	2.20E-07
Internal Events	1.62E-08	5.38E-08	8.08E-08	6.46E-08	2.69E-08
Combined	1.48E-07	4.93E-07	7.39E-07	5.92E-07	2.46E-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 increase due to increasing the interval from 10 to 15 years is 2.46E-7; the total change in LERF due to increasing the ILRT interval from 3 to 15 years is 5.92E-7, which meets the guidance for "small" change in risk, as it exceeds 1.0E-7/yr and remains less than a 1.0E-6 change in LERF. For this change in LERF to be acceptable, total LERF must be less than 1.0E-5. The total LERF due to increasing the ILRT interval from 3 to 15 years is calculated below by adding external and internal events LERF and change in LERF:

$$\mathrm{LERF} = \mathrm{LERF_{IE}} + \mathrm{LERF_{fire}} + \mathrm{LERF_{seismic}} + \mathrm{LERF_{HW}} + \mathrm{LERF_{class3Bincrease}}$$

$$\mathrm{LERF} = 3.20\text{E-}8/\mathrm{yr} + 4.07\text{E-}8/\mathrm{yr} + 3.41\text{E-}6/\mathrm{yr} + 7.98\text{E-}9/\mathrm{yr} + 5.92\text{E-}7/\mathrm{yr} = 4.08\text{E-}6/\mathrm{yr}$$

The total LERF due to increasing the ILRT interval from 10 to 15 years is calculated below by adding external and internal events LERF and change in LERF:

$$\mathrm{LERF} = \mathrm{LERF_{IE}} + \mathrm{LERF_{fire}} + \mathrm{LERF_{seismic}} + \mathrm{LERF_{HW}} + \mathrm{LERF_{class3Bincrease}}$$

$$\mathrm{LERF} = 3.20\text{E-}8/\mathrm{yr} + 4.07\text{E-}8/\mathrm{yr} + 3.41\text{E-}6/\mathrm{yr} + 7.98\text{E-}9/\mathrm{yr} + 2.46\text{E-}7/\mathrm{yr} = 3.73\text{E-}6/\mathrm{yr}$$

Several conservative assumptions were made in this ILRT analysis, as discussed in Sections 4.0, 5.1.3, 5.2.1, and 5.2.4; therefore, the total change in LERF is considered conservative for this application. As specified in Regulatory Guide 1.174 [Reference 4], since the total LERF is less than 1.0E-5, it is acceptable for the ΔLERF to be between 1.0E-7 and 1.0E-6.

5.2.7.1 Other External Hazards

Callaway's Other External Hazards (OEHs) Screening Assessment Notebook [Reference 32] provides an evaluation of OEHs performed in accordance with the process specified in Part 6, Section 6-2 of Reference 48. All the external hazards that could potentially affect the CEC site were screened out based on the criteria specified in SRs EXT-B1 and EXT-C1 of the PRA Standard [Reference 32].

Similarly, Reference 27 analyzed and screened all the hazards listed in Table D-1 of Regulatory Guide 1.200 Revision 3 [Reference 43]. The evaluation concluded that all other hazards either do not present a design-basis challenge to Callaway, the challenge is adequately addressed in the PRA, or the hazard has a negligible impact [Reference 27].

5.2.8 Defense-In-Depth Impact

Regulatory Guide 1.174, Revision 3 [Reference 4] describes an approach that is acceptable for developing risk-informed applications for a licensing basis change that considers engineering issues and applies risk insights. One of the considerations included in RG 1.174 is Defense in Depth. Defense in Depth is a safety philosophy that employs successive compensatory measures to prevent accidents or mitigate damage if a malfunction, accident, or naturally caused event occurs at a nuclear facility. The following seven considerations as presented in RG 1.174, Revision 3, Section C.2.1.1.2 will serve to evaluate the proposed licensing basis change for overall impact on Defense in Depth.

1. <u>Preserve a reasonable balance among the layers of defense.</u>

The use of the risk metrics of LERF, population dose, and conditional containment failure probability collectively ensures the balance between prevention of core damage, prevention of containment failure, and consequence mitigation is preserved. The change in LERF is "very small" with respect to internal events and "small" when including external events per RG 1.174, and the change in population dose and CCFP are "small" as defined in this analysis and consistent with NEI 94-01 Revision 3-A.

2. <u>Preserve adequate capability of design features without an overreliance on programmatic activities as compensatory measures.</u>

The adequacy of the design feature (the containment boundary subject to Type A testing) is preserved as evidenced by the overall "small" change in risk associated with the Type A test frequency change.

3. <u>Preserve system redundancy, independence, and diversity commensurate with the expected frequency and consequences of challenges to the system, including consideration of uncertainty.</u>

The redundancy, independence, and diversity of the containment subject to the Type A test is preserved, commensurate with the expected frequency and consequences of challenges to the system, as evidenced by the overall "small" change in risk associated with the Type A test frequency change.

4. <u>Preserve adequate defense against potential CCFs.</u>

Adequate defense against CCFs is preserved. The Type A test detects problems in the containment which may or may not be the result of a CCF; such a CCF may affect failure of another portion of containment (i.e., local penetrations) due to the same phenomena. Adequate defense against CCFs is preserved via the continued performance of the Type B and C tests and the performance of inspections. The change to the Type A test, which bounds the risk associated with containment failure modes including those involving CCFs, does not degrade adequate defense as evidenced by the overall “small” change in risk associated with the Type A test frequency change.

5. <u>Maintain multiple fission product barriers.</u>

Multiple Fission Product barriers are maintained. The portion of the containment affected by the Type A test extension is still maintained as an independent fission product barrier, albeit with an overall “small” change in the reliability of the barrier.

6. <u>Preserve sufficient defense against human errors.</u>

Sufficient defense against human errors is preserved. The probability of a human error to operate the plant, or to respond to off-normal conditions and accidents is not significantly affected by the change to the Type A testing frequency. Errors committed during test and maintenance may be reduced by the less frequent performance of the Type A test (less opportunity for errors to occur).

7. <u>Continue to meet the intent of the plant’s design criteria.</u>

The intent of the plant’s design criteria continues to be met. The extension of the Type A test does not change the configuration of the plant or the way the plant is operated.

5.2.9 Containment Overpressure

The guidance in Reference 24 states that in general, CDF is not significantly impacted by an extension of the ILRT interval. Plants that rely on containment overpressure for net positive suction head (NPSH) for emergency core coolant system (ECCS) injection for certain accident sequences may experience an increase in CDF.

In the case of accident sequences that are the result of the long-term loss of containment heat removal, containment pressurization and eventual failure are assumed to result in a loss of core coolant injection systems.

In the case where containment overpressure may be a consideration, plants should examine their ECCS NPSH requirements to determine if containment overpressure is required (and assumed to be available) in various accident scenarios. Examples include the following:

- LOCA scenarios where the initial containment pressurization helps to satisfy the NPSH requirements for early injection in BWRs or PWR sump recirculation.
- Total loss of containment heat removal scenarios where gradual containment pressurization helps to satisfy the NPSH requirements for long-term use of an injection system from a source inside of containment (for example, BWR suppression pool).

Either of these scenarios could be impacted by a large containment failure that eliminates the overpressure contribution to the available NPSH calculation. If either of these cases is susceptible to whether or not containment overpressure is available (or other cases are identified), then the PRA model should be adjusted to account for this requirement. As a first-order estimate of the impact, it can be assumed that the EPRI Class 3b contribution would lead to loss of containment overpressure, and the systems that require this contribution to NPSH should be made unavailable when such an isolation failure exists. The impact on CDF can then be accounted for in a similar fashion to the LERF contribution as the EPRI Class 3b contribution changes for various ILRT test intervals. The combined impacts on CDF and LERF should then be considered in the ILRT evaluation and compared with the Regulatory Guide 1.174 acceptance guidelines.

Per the ECCS analysis of record, Callaway does not rely on containment overpressure for net positive suction head (NPSH) for ECCS injection [References 62 and 63].

A separate analysis for Generic Safety Issue (GSI) 191 pertains to debris accumulation in containment following a LOCA [Reference 64]. The Callaway Small LOCA analysis does not credit recirculation to achieve successful core cooling, so overpressure credit for NPSH is not a concern [References 40 and 65]. Reference 40 provides analysis for boiling (flashing) at the top of the sump strainers, deaeration, and NPSH (NPSH is the relevant analysis for this ILRT application). Although spurious or significantly miscalibrated sump level indication is unlikely, these calculations assume a 3-inch penalty to the normal sump level indicator. If the sump level indicator performs nominally, the pool will be 3 inches deeper than the flashing calculation assumes, and no overpressure credit is required. Per Reference 40, “if nominal performance of the normal sump level indicator is assumed (no assumed penalty for a spurious low-level reading), overpressure credit is not required.”

Therefore, the impact of credit for containment overpressure is not significant to the ILRT extension PRA risk metrics, and the conclusion of the application is based on the change in LERF, change in population dose, and change in conditional containment failure probability metrics.

5.3 Sensitivities

5.3.1 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 5.2.6. 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 Table 5-18 – Table 5-20. 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 5-18 – Steel Liner Corrosion Sensitivity Case: 3B Contribution

	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)
Corrosion Likelihood X 1	1.62E-08	5.38E-08	8.08E-08	3.77E-08	6.46E-08	2.69E-08
Corrosion Likelihood X 1000	1.63E-08	5.66E-08	9.05E-08	4.03E-08	7.42E-08	3.39E-08
Corrosion Likelihood X 10000	1.76E-08	8.17E-08	1.78E-07	6.41E-08	1.61E-07	9.66E-08
Corrosion Likelihood X 100000	3.05E-08	3.32E-07	1.06E-06	3.02E-07	1.03E-06	7.24E-07

Table 5-19 – 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)
Corrosion Likelihood X 1	5.14E-01	5.19E-01	5.23E-01	5.33E-03	9.13E-03	3.81E-03
Corrosion Likelihood X 1000	5.14E-01	5.20E-01	5.23E-01	5.37E-03	9.21E-03	3.84E-03
Corrosion Likelihood X 10000	5.14E-01	5.20E-01	5.24E-01	5.80E-03	9.95E-03	4.14E-03
Corrosion Likelihood X 100000	5.16E-01	5.26E-01	5.33E-01	1.01E-02	1.73E-02	7.19E-03

Table 5-20 – 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)
Corrosion Likelihood X 1	4.62E-03	1.54E-02	2.31E-02	1.08E-02	1.85E-02	7.70E-03
Corrosion Likelihood X 1000	4.66E-03	1.62E-02	2.59E-02	1.15E-02	2.12E-02	9.69E-03
Corrosion Likelihood X 10000	5.03E-03	2.34E-02	5.10E-02	1.83E-02	4.60E-02	2.76E-02
Corrosion Likelihood X 100000	8.73E-03	9.51E-02	3.02E-01	8.64E-02	2.93E-01	2.07E-01

5.3.2 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 [Reference 24]. 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 Jeffreys 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 L_a for small and 100 L_a for large) are used here. Table 5-21 presents the magnitudes and probabilities associated with the Jeffreys non-informative prior and the expert elicitation used in the base methodology and this sensitivity case.

Table 5-21 – Callaway Summary of ILRT Extension Using Expert Elicitation Values (from Reference 24)

Leakage Size (L_a)	Expert Elicitation Mean Probability of Occurrence	Percent Reduction
10	3.88E-03	86%
100	2.47E-04	91%

Taking the baseline analysis and using the values provided in Table 5-10 and Table 5-11 for the expert elicitation sensitivity yields the results in Table 5-22.

Table 5-22 – Callaway 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	3.45E-06	3.42E-06	2.86E+03	9.80E-03	3.36E-06	9.60E-03	3.31E-06	9.46E-03
2	7.72E-09	7.72E-09	7.66E+05	5.91E-03	7.72E-09	5.91E-03	7.72E-09	5.91E-03
3a	N/A	2.73E-08	2.86E+04	7.82E-04	9.11E-08	2.61E-03	1.37E-07	3.91E-03
3b	N/A	1.74E-09	2.86E+05	4.98E-04	5.80E-09	1.66E-03	8.70E-09	2.49E-03
7	3.59E-06	3.59E-06	3.19E+05	1.15E+00	3.59E-06	1.15E+00	3.59E-06	1.15E+00
8	2.12E-08	2.12E-08	1.09E+06	2.31E-02	2.12E-08	2.31E-02	2.12E-08	2.31E-02
Totals	7.08E-06	7.08E-06	2.49E+06	1.19E+00	7.08E-06	1.19E+00	7.08E-06	1.19E+00
ΔLERF (3 per 10 yrs base)	N/A				4.06E-09		6.96E-09	
ΔLERF (1 per 10 yrs base)	N/A				N/A		2.90E-09	
CCFP	51.21%				51.26%		51.31%	

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.

6.0 RESULTS

The Internal Events results from this ILRT extension risk assessment for Callaway are summarized in Table 6-1.

Table 6-1 – ILRT Extension Summary (Internal Events)

Class	Dose (person-rem)	Base Case 3 in 10 Years		Extend to 1 in 10 Years		Extend to 1 in 15 Years	
		CDF/yr	Person-Rem/yr	CDF/yr	Person-Rem/yr	CDF/yr	Person-Rem/yr
1	2.86E+03	3.37E-06	9.65E-03	3.18E-06	9.11E-03	3.05E-06	8.72E-03
2	7.66E+05	7.72E-09	5.91E-03	7.72E-09	5.91E-03	7.72E-09	5.91E-03
3a	2.86E+04	6.49E-08	1.86E-03	2.16E-07	6.19E-03	3.25E-07	9.28E-03
3b	2.86E+05	1.62E-08	4.62E-03	5.38E-08	1.54E-02	8.08E-08	2.31E-02
7	3.19E+05	3.59E-06	1.15E+00	3.59E-06	1.15E+00	3.59E-06	1.15E+00
8	1.09E+06	2.12E-08	2.31E-02	2.12E-08	2.31E-02	2.12E-08	2.31E-02
Total		**7.08E-06**	**1.19E+00**	**7.08E-06**	**1.21E+00**	**7.08E-06**	**1.22E+00**
ILRT Dose Rate from 3a and 3b							
ΔTotal Dose Rate (Person-Rem/yr)	From 3 Years	N/A		1.46E-02		2.50E-02	
	From 10 Years	N/A		N/A		1.04E-02	
%ΔDose Rate	From 3 Years	N/A		1.22%		2.10%	
	From 10 Years	N/A		N/A		0.86%	
3b Frequency (LERF/yr)							
ΔLERF	From 3 Years	N/A		3.77E-08		6.46E-08	
	From 10 Years	N/A		N/A		2.69E-08	
CCFP %							
ΔCCFP%	From 3 Years	N/A		0.533%		0.913%	
	From 10 Years	N/A		N/A		0.381%	

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. Regulatory Guide 1.174 defines "small" changes in risk as resulting in increases of CDF greater than 1.0E-6/yr and less than 1.0E-5/yr and increases in LERF greater than 1.0E-7/yr and less than 1.0E-6/yr. 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 6.46E-8/yr using the EPRI guidance; this value increases negligibly if the risk impact of corrosion-induced leakage of the steel liners occurring and going undetected during the extended test interval is included. Therefore, the estimated change in LERF is determined to be "very small" using the acceptance guidelines of Regulatory Guide 1.174 [Reference 4]. The risk change resulting from a change in the Type A ILRT test interval from 3 in 10 years to 1 in 15 years bounds the 1 in 10 years to 1 in 15 years risk change. Considering the increase in LERF resulting from a change in the Type A ILRT test interval from 1 in 10 years to 1 in 15 years is estimated as 2.69E-8/yr, the risk increase is "very small" using the acceptance guidelines of Regulatory Guide 1.174 [Reference 4].
- When external event risk is included, 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 5.92E-7/yr using the EPRI guidance, and total LERF is 4.08E-6/yr. As such, the estimated change in LERF is determined to be "small" using the acceptance guidelines of Regulatory Guide 1.174 [Reference 4]. The risk change resulting from a change in the Type A ILRT test interval from 3 in 10 years to 1 in 15 years bounds the 1 in 10 years to 1 in 15 years risk change. When external event risk is included, the increase in LERF resulting from a change in the Type A ILRT test interval from 1 in 10 years to 1 in 15 years is estimated as 2.46E-7/yr, and the total LERF is 3.73E-6/yr. Therefore, the risk increase is "small" 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 0.025 person-rem/yr. NEI 94-01 [Reference 36] states that a "small" population dose is defined as an increase of ≤ 1.0 person-rem per year, or ≤ 1% of the total population dose, whichever is less restrictive for the risk impact assessment of the extended ILRT intervals. The results of this calculation meet these criteria. Moreover, the risk impact for the ILRT extension when compared to other severe accident risks is negligible.
- The increase in the conditional containment failure probability from the 3 in 10 year interval to 1 in 15 year interval is 0.913%. NEI 94-01 [Reference 1] states that increases in CCFP of ≤ 1.5% is "small." Therefore, this increase is judged to be "small."

Therefore, increasing the ILRT interval to 15 years is considered to be "small" since it represents a small change to the Callaway 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 conclusions for Callaway confirm these general conclusions on a plant-specific basis considering the severe accidents evaluated for Callaway, the Callaway containment failure modes, and the local population surrounding Callaway.

A. PRA ACCEPTABILITY

A.1. Introduction

This appendix provides information on the technical adequacy of the Callaway Plant, Unit No. 1 (Callaway) Probabilistic Risk Assessment (PRA) Internal Events, Internal Flooding, High Winds, Fire, and Seismic PRA models.

Nuclear Energy Institute (NEI) Topical Report NEI 06-09-A, Revision 0 [Reference 45], as clarified by the NRC final safety evaluation of this report [Reference 46], defines the technical attributes of a PRA model and its associated Configuration Risk Management Program (CRMP) tool required to implement this risk-informed application. Meeting these requirements satisfies Regulatory Guide (RG) 1.174, “An Approach for Determining the Technical Adequacy of Probabilistic Risk Assessment Results for Risk-Informed Activities,” Revision 3 [Reference 4], requirements for risk-informed plant-specific changes to a plant's licensing basis.

Ameren Missouri employs a multi-faceted approach to establishing and maintaining the technical adequacy and fidelity of PRA models for Callaway. This approach includes both a PRA maintenance and update process procedure and the use of self-assessments and independent peer reviews.

The Callaway PRA models are at-power models consisting of four hazard models – Internal Flooding, Fire, Seismic, and High Wind. Each hazard model has the Internal Events model as the base with hazard specific initiators added and fault tree modifications and additions made, as necessary. Each model directly addresses plant configurations during plant Modes 1, 2 and 3 of reactor operation. The models provide both core damage frequency (CDF) and large early release frequency (LERF). All five of these PRA models were developed to comply with RG 1.200 Revision 2 [Reference 37].

- Section A.2 describes the peer review findings closure process.
- Section A.3 describes the requirements related to the scope of the Callaway PRA models.
- Section A.4 addresses the technical adequacy of the Callaway PRA Internal Events and Internal Flooding model for this application.
- Section A.5 addresses the technical adequacy of the Callaway PRA High Winds model for this application.
- Section A.6 addresses the technical adequacy of the Callaway PRA Seismic model for this application.
- Section A.7 addresses the technical adequacy of the Callaway PRA Fire model for this application.

A.2. Peer Review Findings Closure Process

All of the PRA models discussed in this appendix have been peer reviewed and assessed against RG 1.200 Revision 2 [Reference 37].

The review and closure of finding-level F&Os was performed by an independent assessment team using the process documented in Appendix X to NEI 05-04, NEI 07-12 and NEI 12-13, "Close-out of Facts and Observations" (F&Os) [Reference 33] as accepted by NRC in the letter dated May 3, 2017 (ML17079A427) [Reference 35]. All of the reviews also met the requirements of NEI 17-07 Revision 2 [Reference 47].

The assessment team assessed whether each F&O was closed through application of a PRA maintenance or upgrade activity, as defined by the ASME/ANS PRA Standard, or through application of a new method. Note that, per APC 17-13, Subject: “NRC Acceptance of Industry

Guidance on Closure of PRA Peer Review Findings," dated May 8, 2017 with attachment Appendix X, a new method represents a fundamentally new approach in addressing a technical aspect of PRA. The results of the peer reviews and independent assessments have been documented and are available for NRC audit.

The PRA scope and technical adequacy is met for this application as the Standard requirements for all models are met at Capability Category II (CCII) or higher. There are no open Finding F&Os against any of the models discussed in this application, and all Finding F&Os have been independently assessed and closed using the processes discussed above. The resolved findings and the basis for resolution are documented in the Callaway PRA documentation and the F&O Closure Review reports.

A.3. Scope of the Callaway PRA Models

The Internal Events, Internal Flooding, Fire, High Winds, and Seismic PRA models are at-power models (i.e., they directly address plant configurations during plant Modes 1, 2 and 3 of reactor operation). The models provide both core damage frequency (CDF) and large early release frequency (LERF).

Note that the Callaway PRA models do not incorporate the risk impacts of external events except for High Winds and Seismic. The treatment of non-modeled external risk hazards are discussed in the OEH Notebook [Reference 32] and Enclosure 4 [Reference 27], which show that all non-modeled external risk hazards screen.

A.4. Technical Adequacy of the Callaway Internal Events and Internal Flooding PRA Model

Topical Report NEI 06-09-A requires that the PRA be reviewed to the guidance of RG 1.200 [Reference 37] for a PRA that meets Capability Category II (CCII) for the supporting requirements of the Internal Events at power ASME/ANS PRA Standard [Reference 48].

The information provided in this section demonstrates that the Callaway Internal Events PRA model (including Internal Flooding) meets the expectations for PRA scope and technical adequacy as presented in ASME/ANS RA-Sa-2009 [Reference 48] and RG 1.200 to fully support this ILRT extension application. The Ameren Missouri risk management process ensures that the PRA model used in this application reflects the as-built and as-operated plant for Callaway.

Related to the technical adequacy of the Internal Events model, the Internal Events discussion below describes implementation of the methodology provided in PWROG-18027-NP [Reference 49] for assessing the loss of room cooling in PRA modeling. Following, but unrelated to, implementation of the method provided in PWROG-18027-NP into the Callaway PRA, this method was chosen by the PWROG and NEI to pilot the Newly Developed Methods (NDM) peer review process established in NEI 17-07 [Reference 47]. The NEI 17-07 process was successfully completed with all applicable NDM attributes met at Capability Category I/II (CC I/II) and no open peer review Findings against the method in PWROG-18027-NP.

In addition, an implementation peer review and associated F&O closure review have been completed using NRC-approved processes, with no open Findings identified against implementation of the method. While the NEI 17-07 process was completed successfully, it is recognized that this process was not an endorsed process until RG 1.200, "Acceptability of Probabilistic Risk Assessment Results for Risk-Informed Activities," Revision 3 was issued in December 2020. As a result, the NRC staff may decide to independently review the method in PWROG-18027-NP for technical adequacy. The PWROG-18027-NP report contains the technical basis for the acceptability of the method and is available for NRC audit.

Peer Review Summary

The Internal Events/Internal Flooding PRA was peer reviewed in April 2019. This peer review was a full-scope review of the technical elements of the Internal Events and Internal Flooding at-power PRA as documented in PWROG-19012-P [Reference 50]. As a full scope review, it included those supporting requirements (SRs) specified in PWROG-19020-NP [Reference 44] for implementation of the methodology for loss of room cooling modeling provided in PWROG-18027-NP [Reference 49].

An Independent Assessment of F&Os was conducted in November 2019 and documented in PWROG-19034-P [Reference 51]. The scope of the assessment included all Facts and Observations (F&Os) generated in the April 2019 peer review. All F&Os except for one were closed. The remaining F&O was related to implementation of the methodology provided in PWROG-18027-NP [Reference 49] for assessing the loss of room cooling in PRA modeling. Following, but unrelated to, incorporation of the method provided in PWROG-18027-NP into the Callaway PRA, this method was chosen by the PWROG and NEI to pilot the Newly Developed Methods (NDM) peer review process established in NEI 17-07 [Reference 47]. Despite the Callaway assessment, and acknowledgement by the PWROG, that the method provided in PWROG-18027-NP did not necessarily meet the definition of a NDM, Callaway decided to suspend resolution of the associated F&O until the NDM peer review and closure of any F&Os were completed using the process established in NEI 17-07. Also, during the November 2019 independent assessment, two F&O resolutions were determined to be upgrades to the Internal Events/Internal Flooding PRA. Thus, a focused-scope peer review was required. Based on this focused scope peer review, one new Internal Events F&O was generated.

During February and March 2020, a new peer review, following the guidance in NEI 17-07 Revision 2, was conducted on the method provided in PWROG-18027-NP and documented in PWROG-19020-NP. Based on the results of this review all applicable NDM attributes are met at CC I/II and there are no open peer review Findings against the method in PWROG-18027-NP.

In June 2020, an independent assessment of F&O resolution and a focused scope peer review, completing the review of PWROG-18027-NP implementation, were conducted on the Callaway Internal Events and Fire PRA models. The focused scope peer review determined that all of the SRs that were examined, including the SR associated with the F&O related to implementation of the method in PWROG-18027-NP, satisfy CCII or higher requirements as documented in AMN#PES00031-REPT-001 [Reference 52]. The independent assessment of F&Os included an assessment of all remaining open F&O Findings. The results of this review are documented in AMN#PES00031-REPT-002 [Reference 53].

There are no open peer review Findings for the Internal Events/Internal Flooding PRA model.

A.5. Technical Adequacy of Callaway High Winds PRA Model

The information provided in this section demonstrates that the Callaway High Winds PRA model meets the expectations for PRA scope and technical adequacy as presented in ASME/ANS RA-Sa-2009 [Reference 48] and RG 1.200 to fully support this ILRT extension application. The Ameren Missouri risk management process ensures that the PRA model used in this application reflects the as-built and as-operated plant for Callaway.

Peer Review Summary

The High Winds PRA was peer reviewed in April 2019 and documented in PWROG-19022-P [Reference 54]. The scope of this work was to review the Callaway External Hazards Screening Assessment and High Winds PRA against the technical elements in Sections 6 and 7 of the ASME/ANS RA-Sa-2009 Standard, and in RG 1.200.

An Independent Assessment of F&O resolution was conducted in November 2019 and documented in PWROG-19034-P [Reference 51]. The scope of the assessment included all F&Os generated in the April 2019 peer review. All F&Os were closed.

There are no open peer review Findings for the Other External Hazards Screening or the High Winds PRA model.

A.6. Technical Adequacy of Callaway Seismic PRA Model

The information provided in this section demonstrates that the Callaway Seismic PRA model meets the expectations for PRA scope and technical adequacy as presented in ASME/ANS RA-S CASE 1, Case for ASME/ANS RA-Sb-2013 [Reference 55] and RG 1.200 to fully support this ILRT extension application. The Ameren Missouri risk management process ensures that the PRA model used in this application reflects the as-built and as-operated plant for Callaway.

<u>Peer Review Summary</u>

The Seismic PRA was peer reviewed in June 2018 and documented in PWROG-18044-P [Reference 56]. This peer review was conducted against the requirements of the Code Case for ASME/ANS RA-Sb-2013 [Reference 55], as amended by the Nuclear Regulatory Commission (NRC) on March 12, 2018 [Reference 57]. The Code Case is an approved alternative to Part 5 of ASME/ANS RA-Sb-2013 Addendum B, the American Society of Mechanical Engineers (ASME) / American Nuclear Society (ANS) Probabilistic Risk Assessment (PRA) Standard.

An Independent Assessment of F&Os was conducted in March 2019. The scope of the assessment included all but two of the F&Os generated in the June 2018 peer review. All in-scope F&Os were closed as documented in PWROG-19011-P [Reference 58]. Also, in the March 2019 review documented in PWROG-19011-P, three SRs were the subject of a focused-scope peer review based on the closures of associated F&Os being assessed as upgrades. As a result of that peer review, the three SRs were determined to be met at CCII.

Subsequently, another Independent Assessment of F&Os was conducted in June 2020 and documented in AMN#PES00031-REPT-002 [Reference 53]. The scope of the assessment included all remaining F&Os generated in the June 2018 peer review. All F&Os were closed.

There are no open peer review Findings for the Seismic PRA model.

A.7. Technical Adequacy of Callaway Fire PRA Model

The information provided in this section demonstrates that the Callaway Fire PRA model meets the expectations for PRA scope and technical adequacy as presented in ASME/ANS RA-Sa-2009 [Reference 48] and RG 1.200 to fully support this ILRT extension application. The Ameren Missouri risk management process ensures that the PRA model used in this application reflects the as-built and as-operated plant for Callaway.

The Internal Fire PRA model was developed consistent with NUREG/CR-6850 [Reference 59] and only utilizes methods previously accepted by the NRC. Callaway was approved to implement NFPA-805 in January 2014, and since that time, there have been numerous updates to the approved methods through the issuance of Fire PRA frequently asked questions and new or revised guidance documents. New or revised guidance is specifically addressed through the Callaway PRA maintenance and update process. The Ameren Missouri risk management process ensures that the PRA model used in this application reflects the as-built and as-operated plant for Callaway.

It should also be noted that, as part of transition to NFPA 805, there were several committed modifications and implementation items as documented in NFPA 805 LAR Attachment S, "Plant

Modifications and Items to be Completed during Implementation," which described the Callaway plant modifications necessary to implement the NFPA 805 licensing basis. All NFPA 805 LAR Attachment S items have been implemented; therefore, there are no NFPA 805 open items impacting this application.

Peer Review Summary

The Fire PRA was prepared using the methodology defined in NUREG/CR-6850, "Fire PRA Methodology for Nuclear Power Facilities," to support a transition to National Fire Protection Association (NFPA) Standard 805, "Performance Based Standard for Fire Protection for Light Water Reactor Electric Generating Plants." The Fire PRA was peer reviewed to ASME/ANS RA-Sa-2009 and RG 1.200 Revision 2 in October 2009. The review is documented in LTR-RAM-11-10-019 [Reference 60].

An Independent Assessment of F&Os was conducted in June 2019 and documented in AMN#PES00021-REPT-001 [Reference 61].

In June 2020, an independent assessment of F&Os and a focused scope peer review were conducted for the Callaway Internal Events and Fire PRA models. The focused scope peer review generated additional Fire PRA related F&Os as documented in AMN#PES00031-REPT-001 [Reference 52]. The independent assessment of F&Os included an assessment of all remaining open F&O Findings. As documented in AMN#PES00031-REPT-002 [Reference 53], all Finding F&Os were closed, including the Fire PRA Findings identified in the Focused Scope peer review.

In fulfillment of Commitment 50437 in Enclosure 4 to ULNRC-06550 (ML20304A456) and associated with closure of NFPA 805 LAR Table S-3 Implementation Item 13-805-001, a focused scope peer review was conducted in November 2020, as documented in AMN#PES00031-REPT-003 [Reference 29], for the resolution of Fire PRA Suggestion F&O FSS-B1-03, which a July 2019 F&O closure review had determined to be an upgrade, as documented in AMN#PES00021-REPT-001 [Reference 61].

As documented in AMN#PES00042-REPT-002 [Reference 26], the F&Os from this focused scope peer review were closed during an F&O closure review in February 2021. The results of this review formally closed Commitment 50437.

There are no open peer review Findings for the Fire PRA model.

A.8. Summary

The PRA scope and technical adequacy is met for this application as the Standard requirements for all models are met at CCII or higher. There are no open Finding F&Os against any of the models discussed in this Enclosure, and all Finding F&Os have been independently assessed and closed using the processes discussed in Section 2 of this Enclosure. In addition, all of the reviews also met the requirements of NEI 17-07 Revision 2 [Reference 47].