ATTACHMENT 3a
Evaluation of Risk Significance of Permanent ILRT Extension

Braidwood Station, Units 1 and 2

Renewed Facility Operating License Nos. NPF-72 and NPF-77

Calculation 54017-CALC-01

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Advancing the Science of Safety

Braidwood Station:
Evaluation of Risk Significance of Permanent ILRT Extension

54017-CALC-01

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REVISION RECORD SUMMARY

Revision	Revision Summary
0	Initial Issue

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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 Braidwood Station (BRW). The risk assessment follows the guidelines from NEI 94-01, Revision 3-A [Reference 1], the NEI "Interim Guidance for Performing Risk Impact Assessments in Support of One-Time Extensions for Containment Integrated Leakage Rate Test Surveillance Intervals" from November 2001 [Reference 3], the NRC regulatory guidance on the use of Probabilistic Risk Assessment (PRA) as stated in Regulatory Guide 1.200 as applied to ILRT interval extensions, 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 BRW.

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. This methodology is followed to determine the appropriate risk information for use in evaluating the impact of the proposed ILRT changes.

It should be noted that containment leak-tight integrity is also verified through periodic in-service

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

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

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

In addition, the total annual risk (person-rem/year population dose) is examined to demonstrate the relative change in this parameter. While no acceptance guidelines for these additional figures of merit are published, examinations of NUREG-1493 and Safety 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/year and/or 0.002% to 0.46% of the total accident dose. The total doses for the spectrum of all accidents (NUREG-1493 [Reference 6], Figure 7-2) result in health effects that are at least two orders of magnitude less than the NRC Safety Goal Risk. Given these perspectives, a "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. *Revision 3-A to Industry Guideline for Implementing Performance-Based Option of 10 CFR Part 50, Appendix J*, NEI 94-01, July 2012.
2. *Risk Impact Assessment of Revised Containment Leak Rate Testing Intervals*, EPRI, Palo Alto, CA EPRI TR-104285, August 1994.
3. *Interim Guidance for Performing Risk Impact Assessments in Support of One-Time Extensions for Containment Integrated Leakage Rate Test Surveillance Intervals*, Revision 4, developed for NEI by EPRI and Data Systems and Solutions, November 2001.
4. An Approach for Using Probabilistic Risk Assessment in Risk-Informed Decisions on Plant-Specific Changes to the Licensing Basis, Regulatory Guide 1.174, 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. Calculation BB-ASM-007, Byron / Braidwood PRA Application Notebook, "Application Specific Model Notebook," Revision 0, December 2018.

18. Calculation BB-ASM-005, Byron and Braidwood Probabilistic Risk Assessment, "Application-Specific Model (ASM)," Revision 1, March 2019.
19. Braidwood Station Environmental Report, Appendix F, "Severe Accident Mitigation Alternatives Analysis," Revision 2.
20. Anthony R. Pietrangelo, One-time extensions of containment integrated leak rate test interval – additional information, NEI letter to Administrative Points of Contact, November 30, 2001.
21. Letter from J. A. Hutton (Exelon, Peach Bottom) to U. S. Nuclear Regulatory Commission, Docket No. 50-278, License No. DPR-56, LAR-01-00430, dated May 30, 2001.
22. *Risk Assessment for Joseph M. Farley Nuclear Plant Regarding ILRT (Type A) Extension Request*, prepared for Southern Nuclear Operating Co. by ERIN Engineering and Research, P0293010002-1929-030602, March 2002.
23. Letter from D. E. Young (Florida Power, Crystal River) to U. S. Nuclear Regulatory Commission, 3F0401-11, dated April 25, 2001.
24. *Risk Impact Assessment of Extended Integrated Leak Rate Testing Intervals*, Revision 2-A of 1009325, EPRI, Palo Alto, CA, 1018243, October 2008.
25. *Risk Assessment for Vogtle Electric Generating Plant Regarding the ILRT (Type A) Extension Request*, prepared for Southern Nuclear Operating Co. by ERIN Engineering and Research, February 2003.
26. Perspectives Gained from the IPEEE Program, USNRC, NUREG-1742, April 2002.
27. Byron/Braidwood Nuclear Stations, "Updated Final Safety Analysis Report," Revision 14, December 2012.
28. Calculation BB-MISC-028, "External Hazards Assessment for Byron and Braidwood Stations," Revision 4, November 2018.
29. U.S. Nuclear Regulatory Commission and EPRI, "EPRI/NRC-RES Fire PRA Methodology for Nuclear Power Facilities," in EPRI 1011089 - NUREG/CR-6850, August 2005.
30. Work Order Package 99200555 01, Braidwood Generating Station, "ILRT Shall be Conducted in Accordance with Appdx J," September 16, 2013.
31. Procedure 1BwVSR 3.6.1.1.ILRT, "Unit 1 Primary Containment Type A Integrated Leakage Rate Test (ILRT)," Revision 12.
32. Procedure 2BwVSR 3.6.1.1.ILRT, "Unit 2 Primary Containment Type A Integrated Leakage Rate Test (ILRT)," Revision 9.
33. Report 032299-RPT-05, "Byron Braidwood Nuclear Power Plants; PRA Finding Level Fact and Observation Technical Review," Revision 2, May 2018.
34. Technical Letter Report ML112070867, Containment Liner Corrosion Operating Experience Summary, Revision 1, August 2011.
35. NUREG-0800, Section 6.5.2, "Containment Spray as a Fission Product Cleanup System," Revision 3.

36. Regulatory Guide 1.200, "An Approach for Determining the Technical Adequacy of Probabilistic Risk Assessment Results for Risk-Informed Activities," Revision 2, March 2009.

37. ASME/ANS RA-Sa–2009, Standard for Level 1/Large Early Release Frequency Probabilistic Risk Assessment for Nuclear Power Plant Applications," 2009.

38. NEI 05-04, "Process for Performing Internal Events PRA Peer Reviews Using the ASME/ANS PRA Standard," Revision 2, November 2008.

39. ACUBE 2.0 Software Manual, EPRI Report 3002003169, December 2014.

40. NEI Letter to USNRC, "Final Revision of Appendix X to NEI 05-04/07-12/12-16, Close-Out of Facts and Observations (F&Os)," February 21, 2017, Accession Number ML17086A431.

41. USNRC 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.

4.0 ASSUMPTIONS AND LIMITATIONS

The following assumptions were used in the calculation:

- The acceptability (i.e., technical adequacy) of the BRW PRA [Reference 17] is either consistent with the requirements of Regulatory Guide 1.200, or where gaps exist, the gaps have been addressed, as detailed in Appendix A.
- The BRW Level 1 and 2 internal events PRA models provide representative results.
- It is appropriate to use the BRW 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. The Seismic risk from the External Hazard Assessment [Reference 28] and Fire application specific model (ASM) [Reference 18] are used for this sensitivity analysis.
- 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.
- 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, rounding differences may result in table summations.

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 BRW. 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 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 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..."

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

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

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

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

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

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

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

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

The approach included in this guidance document is used in the BRW 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 BRW ILRT Extension Risk Assessment includes the following:

- CDF and LERF Model results [Reference 17, Reference 18, Reference 28]
- Dose within a 50-mile radius [Reference 19]
- ILRT results to demonstrate adequacy of the administrative and hardware issues [References 29 and 30]

BRW Model

The Internal Events PRA Model that is used for BRW is characteristic of the as-built plant. The current Level 1, LERF, and Level 2 model is a linked fault tree model [Reference 17]. The CDF is 1.24E-5/year for Unit 1 and 1.24E-5/year for Unit 2; the LERF is 7.24E-7/year for Unit 1 and 7.21E-7/year for Unit 2 [Reference 17]. Table 5-1 and Table 5-2 provide a summary of the Internal Events CDF and LERF results for the BRW PRA Model.

The total Fire CDF is 5.62E-5/year for Unit 1 and 5.67E-5/year for Unit 2; the total Fire LERF is 5.39E-6/year for Unit 1 and 5.18E-6/year for Unit 2 [Reference 18]. The High Winds CDF is less than 1E-6/year [Reference 28]. The total Seismic CDF is 4.2E-6/year and the total Seismic LERF is 6.2E-6/year [Reference 28]. 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	Unit 1 Frequency (per year)	Unit 2 Frequency (per year)
Internal Floods	1.80E-06	1.77E-06
Transients	7.31E-06	7.30E-06
Feedwater/Main Steam Break Inside Containment	1.77E-07	1.77E-07
LOCAs	2.14E-06	2.14E-06
ISLOCA	4.44E-08	4.45E-08
SGTR	5.09E-07	5.09E-07
RPV Rupture	2.90E-08	2.90E-08
Loss of Offsite Power (LOOP)	4.24E-07	4.29E-07
Total Internal Events CDF	**1.24E-05**	**1.24E-05**

Table 5-2 – Internal Events LERF

Internal Events	Unit 1 Frequency (per year)	Unit 2 Frequency (per year)
Internal Floods	1.35E-07	1.34E-07
Transients	3.72E-07	3.70E-07
Feedwater/Main Steam Break Inside Containment	1.36E-08	1.36E-08
LOCAs	1.33E-07	1.33E-07
ISLOCA	4.54E-08	4.54E-08
SGTR	2.26E-08	2.27E-08
RPV Rupture	9.41E-11	9.37E-11
LOOP	1.93E-09	1.94E-09
Total Internal Events LERF	**7.24E-07**	**7.21E-07**

Population Dose Calculations

The population dose calculation was reported in the SAMA [Reference 19]. Table 5-3 presents dose exposures calculated from methodology described in Reference 1 and data from Reference 19. 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, five containment end-states correspond to EPRI Accident Class 7 (LERF-CFE, LATE-BMT-AFW, LATE-BMT-NOAFW, LATE-CHR-AFW and LATE-CHR-NOAFW Release Categories); the EPRI Accident Class 7 dose is calculated via a weighted average using the frequencies provided in Reference 19. The SGTR Release Categories (LERF-SGTR-AFW, LERF-SGTR-NOAFW, LERF-ISGTR, SERF-SGTR-TISGTR-HLF, and SERF-SGTR-AFW-SC; SGTR dose is calculated via a weighted average using the frequencies provided in Reference 19) and "LERF-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 represented as $10L_a$ and $100L_a$, respectively, as guidance in Reference 1 dictates.

Table 5-3 – Population Dose

EPRI Category	Dose (person-rem)
Class 1	2.48E+04
Class 2	3.25E+06
Class 7	3.99E+06
Class 8 (SGTR)	5.01E+06
Class 8 (ISLOCA)	4.57E+07

Release Category Definitions

Table 5-4 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 5-4 – 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 5-4, 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 BRW, 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 IP3 request for a one-time ILRT extension that was approved by the NRC [Reference 9]) because it does not factor in the possibility that the failures could be detected by other tests (e.g., the Type B local leak rate tests that will still occur). Eliminating this possibility conservatively over-estimates the factor increases attributable to the ILRT extension.

5.2 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 5-5.

The analysis performed examined BRW-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 5-5 – 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 5-5.

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 5-5 were developed for Braidwood by first determining the frequencies for Classes 1, 2, 6, 7, and 8.

Table 5-6 presents the grouping of each release category in EPRI Classes based on the associated description. Table 5-7 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 were a total of 217 successful ILRTs during this data collection period. Therefore, the probability of leakage is determined for Class 3a as shown in the following equation:

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

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

$$Freq_{U1class3a} = P_{class3a} * (CDF - LERF) = \frac{2}{217} * (1.24\text{E-}5 - 7.24\text{E-}7) = 1.08\text{E-}7$$

$$Freq_{U2class3a} = P_{class3a} * (CDF - LERF) = \frac{2}{217} * (1.24\text{E-}5 - 7.21\text{E-}7) = 1.08\text{E-}7$$

In the database of 217 ILRTs, there are zero containment leakage events that could result in a large early release. Therefore, the Jeffreys non-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:

$$Freq_{U1class3b} = P_{class3b} * (CDF - LERF) = \frac{.5}{218} * (1.24\text{E-}5 - 7.24\text{E-}7) = 2.68\text{E-}8$$

$$Freq_{U2class3b} = P_{class3b} * (CDF - LERF) = \frac{.5}{218} * (1.24\text{E-}5 - 7.21\text{E-}7) = 2.68\text{E-}8$$

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 Intact frequency for internal events is 5.28E-6 for Unit 1 and 5.29E-6 for Unit 2 [Reference 17]. The EPRI Accident Class 1 frequency is then adjusted by subtracting the EPRI Class 3a and 3b frequency (to preserve total CDF), calculated below:

$$Freq_{U1class1} = Freq_{U1Intact} - (Freq_{U1class3a} - Freq_{U1class3b})$$

$$Freq_{U2class1} = Freq_{U2Intact} - (Freq_{U2class3a} - Freq_{U2class3b})$$

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 0.799 of LERF for Unit 1 and 0.796 of LERF for Unit 2. Multiplying by the respective units' LERF, the

EPRI Accident Class 2 frequency is 5.78E-7 for Unit 1 and 5.74E-7 for Unit 2, as shown in Table 5-6.

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 5-6.

Class 8 Sequences. This group consists of all core damage accident progression bins in which containment is bypassed via ISLOCA or SGTR. The ISLOCA initiators are in internal events cutsets that contribute 0.0627 of LERF for Unit 1 and 0.0629 of LERF for Unit 2. The SGTR initiator is in internal events cutsets that contribute 0.0410 of CDF for Unit 1 and 0.0410 of CDF for Unit 2. Thus, the total EPRI Accident Class 8 frequency is the summation of the ISLOCA and SGTR frequencies, 5.54E-7 for Unit 1 and 5.54E-07 for Unit 2, as shown in Table 5-6 and Table 5-7.

Table 5-6 – Accident Class Frequencies

EPRI Category	Unit 1 Frequency (/yr)	Unit 2 Frequency (/yr)
Class 1	6.74E-06	6.75E-06
Class 2	5.78E-07	5.74E-07
Class 7	4.56E-06	4.52E-06
Class 8 (SGTR)	5.09E-07	5.09E-07
Class 8 (ISLOCA)	4.54E-08	4.54E-08
Total (CDF)	1.24E-05	1.24E-05

Table 5-7 – Baseline Risk Profile

Class	Description	Unit 1 Frequency (/yr)	Unit 2 Frequency (/yr)
1	No containment failure	6.60E-06[2]	6.62E-06[2]
2	Large containment isolation failures	5.78E-07	5.74E-07
3a	Small isolation failures (liner breach)	1.08E-07	1.08E-07
3b	Large isolation failures (liner breach)	2.68E-08	2.68E-08
4	Small isolation failures - failure to seal (type B)	ε[1]	ε[1]
5	Small isolation failures - failure to seal (type C)	ε[1]	ε[1]
6	Containment isolation failures (dependent failure, personnel errors)	ε[1]	ε[1]
7	Severe accident phenomena induced failure (early and late)	4.56E-06	4.52E-06
8	Containment bypass	5.54E-07	5.54E-07
	Total	1.24E-05	1.24E-05

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. Table 5-3 provides population dose for each release category. Table 5-8 provides a correlation of BRW population dose to EPRI Accident Class. The population dose for EPRI Accident Classes 3a and 3b were calculated based on the guidance provided in EPRI Report No. 1009325, Revision 2-A [Reference 24] as follows:

$$EPRI\ Class\ 3a\ Population\ Dose = 10 * 2.48E{+}4 = 2.48E{+}5$$

$$EPRI\ Class\ 3b\ Population\ Dose = 100 * 2.48E{+}4 = 2.48E{+}6$$

Table 5-8 – Baseline Population Doses

Class	Description	Unit 1 Population Dose (person-rem)	Unit 2 Population Dose (person-rem)
1	No containment failure	2.48E+04	2.48E+04
2	Large containment isolation failures	3.25E+06	3.25E+06
3a	Small isolation failures (liner breach)	2.48E+05[1]	2.48E+05[1]
3b	Large isolation failures (liner breach)	2.48E+06[2]	2.48E+06[2]
4	Small isolation failures - failure to seal (type B)	N/A	N/A
5	Small isolation failures - failure to seal (type C)	N/A	N/A
6	Containment isolation failures (dependent failure, personnel errors)	N/A	N/A
7	Severe accident phenomena induced failure (early and late)	3.99E+06	3.99E+06
8	Containment bypass	8.34E+06	8.34E+06

1. 10*L_a
2. 100*L_a

Table 5-9 – Unit 1 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]	6.60E-06	53.12%	2.48E+04	1.64E-01
2	Large containment isolation failures	5.78E-07	4.65%	3.25E+06	1.88E+00
3a	Small isolation failures (liner breach)	1.08E-07	0.87%	2.48E+05	2.68E-02
3b	Large isolation failures (liner breach)	2.68E-08	0.22%	2.48E+06	6.66E-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)	4.56E-06	36.68%	3.99E+06	1.82E+01
8	Containment bypass	5.54E-07	4.46%	8.34E+06	4.63E+00
	Total	1.24E-05			2.50E+01

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 in order to preserve total CDF.

Table 5-10 – Unit 2 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]	6.62E-06	53.36%	2.48E+04	1.64E-01
2	Large containment isolation failures	5.74E-07	4.63%	3.25E+06	1.86E+00
3a	Small isolation failures (liner breach)	1.08E-07	0.87%	2.48E+05	2.67E-02
3b	Large isolation failures (liner breach)	2.68E-08	0.22%	2.48E+06	6.64E-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)	4.52E-06	36.45%	3.99E+06	1.81E+01
8	Containment bypass	5.54E-07	4.47%	8.34E+06	4.62E+00
	Total	1.24E-05			2.48E+01

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 in order 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 3-year interval (i.e., a simplified representation of a 3-to-10 interval).

<u>Risk Impact Due to 10-Year Test Interval</u>

As previously stated, Type A tests impact only Class 3 sequences. For Class 3 sequences, the release magnitude is not impacted by the change in test interval (a small or large breach remains the same, even though the probability of not detecting the breach increases). Thus, only the frequency of Class 3a and Class 3b sequences is impacted. The risk contribution is changed based on the NEI guidance as described in Section 5.1.3 by a factor of 10/3 compared to the base case values. The Class 3a and 3b frequencies are calculated as follows:

$$Freq_{U1Class3a10yr} = \frac{10}{3} * \frac{2}{217} * (CDF - LERF) = \frac{10}{3} * \frac{2}{217} * 1.17\text{E-}5 = 3.60\text{E-}7$$

$$Freq_{U1Class3b10yr} = \frac{10}{3} * \frac{.5}{218} * (CDF - LERF) = \frac{10}{3} * \frac{.5}{218} * 1.17\text{E-}5 = 8.95\text{E-}8$$

$$Freq_{U2Class3a10yr} = \frac{10}{3} * \frac{2}{217} * (CDF - LERF) = \frac{10}{3} * \frac{2}{217} * 1.17\text{E-}5 = 3.59\text{E-}7$$

$$Freq_{U2Class3b10yr} = \frac{10}{3} * \frac{.5}{218} * (CDF - LERF) = \frac{10}{3} * \frac{.5}{218} * 1.17\text{E-}5 = 8.93\text{E-}8$$

The results of the calculation for a 10-year interval are presented in Table 5-11.

Table 5-11 – Unit 1 Risk Profile for Once in 10 Year ILRT

Class	Description	Frequency (/yr)	Contribution (%)	Population Dose (person-rem)	Population Dose Rate (person-rem/yr)
1	No containment failure[2]	6.29E-06	50.59%	2.48E+04	1.56E-01
2	Large containment isolation failures	5.78E-07	4.65%	3.25E+06	1.88E+00
3a	Small isolation failures (liner breach)	3.60E-07	2.89%	2.48E+05	8.92E-02
3b	Large isolation failures (liner breach)	8.95E-08	0.72%	2.48E+06	2.22E-01
4	Small isolation failures - failure to seal (type B)	ε[1]	ε[1]	ε[1]	ε[1]
5	Small isolation failures - failure to seal (type C)	ε[1]	ε[1]	ε[1]	ε[1]
6	Containment isolation failures (dependent failure, personnel errors)	ε[1]	ε[1]	ε[1]	ε[1]
7	Severe accident phenomena induced failure (early and late)	4.56E-06	36.68%	3.99E+06	1.82E+01
8	Containment bypass	5.54E-07	4.46%	8.34E+06	4.63E+00
	Total	1.24E-05			2.52E+01

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 in order to preserve total CDF.

Table 5-12 – Unit 2 Risk Profile for Once in 10 Year ILRT

Class	Description	Frequency (/yr)	Contribution (%)	Population Dose (person-rem)	Population Dose Rate (person-rem/yr)
1	No containment failure[2]	6.30E-06	50.83%	2.48E+04	1.56E-01
2	Large containment isolation failures	5.74E-07	4.63%	3.25E+06	1.86E+00
3a	Small isolation failures (liner breach)	3.59E-07	2.89%	2.48E+05	8.90E-02
3b	Large isolation failures (liner breach)	8.93E-08	0.72%	2.48E+06	2.21E-01
4	Small isolation failures - failure to seal (type B)	ε[1]	ε[1]	ε[1]	ε[1]
5	Small isolation failures - failure to seal (type C)	ε[1]	ε[1]	ε[1]	ε[1]
6	Containment isolation failures (dependent failure, personnel errors)	ε[1]	ε[1]	ε[1]	ε[1]
7	Severe accident phenomena induced failure (early and late)	4.52E-06	36.45%	3.99E+06	1.81E+01
8	Containment bypass	5.54E-07	4.47%	8.34E+06	4.62E+00
	Total	1.24E-05			2.50E+01

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 in order 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:

$$Freq_{U1Class3a15yr} = \frac{15}{3} * \frac{2}{217} * (CDF - LERF) = 5 * \frac{2}{217} * 1.17\text{E-}5 = 5.39\text{E-}7$$

$$Freq_{U1Class3b15yr} = \frac{15}{3} * \frac{.5}{218} * (CDF - LERF) = 5 * \frac{.5}{218} * 1.17\text{E-}5 = 1.34\text{E-}7$$

$$Freq_{U2Class3a15yr} = \frac{15}{3} * \frac{2}{217} * (CDF - LERF) = 5 * \frac{2}{217} * 1.17\text{E-}5 = 5.38\text{E-}7$$

$$Freq_{U2Class3b15yr} = \frac{15}{3} * \frac{.5}{218} * (CDF - LERF) = 5 * \frac{.5}{218} * 1.17\text{E-}5 = 1.34\text{E-}7$$

The results of the calculation for a 15-year interval are presented in Table 5-13.

Table 5-13 – Unit 1 Risk Profile for Once in 15 Year ILRT

Class	Description	Frequency (/yr)	Contribution (%)	Population Dose (person-rem)	Population Dose Rate (person-rem/yr)
1	No containment failure[2]	6.06E-06	48.79%	2.48E+04	1.50E-01
2	Large containment isolation failures	5.78E-07	4.65%	3.25E+06	1.88E+00
3a	Small isolation failures (liner breach)	5.39E-07	4.34%	2.48E+05	1.34E-01
3b	Large isolation failures (liner breach)	1.34E-07	1.08%	2.48E+06	3.33E-01
4	Small isolation failures - failure to seal (type B)	ε[1]	ε[1]	ε[1]	ε[1]
5	Small isolation failures - failure to seal (type C)	ε[1]	ε[1]	ε[1]	ε[1]
6	Containment isolation failures (dependent failure, personnel errors)	ε[1]	ε[1]	ε[1]	ε[1]
7	Severe accident phenomena induced failure (early and late)	4.56E-06	36.68%	3.99E+06	1.82E+01
8	Containment bypass	5.54E-07	4.46%	8.34E+06	4.63E+00
	Total	1.24E-05			2.53E+01

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 in order to preserve total CDF.

Table 5-14 – Unit 2 Risk Profile for Once in 15 Year ILRT

Class	Description	Frequency (/yr)	Contribution (%)	Population Dose (person-rem)	Population Dose Rate (person-rem/yr)
1	No containment failure[2]	6.08E-06	49.03%	2.48E+04	1.51E-01
2	Large containment isolation failures	5.74E-07	4.63%	3.25E+06	1.86E+00
3a	Small isolation failures (liner breach)	5.38E-07	4.34%	2.48E+05	1.33E-01
3b	Large isolation failures (liner breach)	1.34E-07	1.08%	2.48E+06	3.32E-01
4	Small isolation failures - failure to seal (type B)	ε[1]	ε[1]	ε[1]	ε[1]
5	Small isolation failures - failure to seal (type C)	ε[1]	ε[1]	ε[1]	ε[1]
6	Containment isolation failures (dependent failure, personnel errors)	ε[1]	ε[1]	ε[1]	ε[1]
7	Severe accident phenomena induced failure (early and late)	4.52E-06	36.45%	3.99E+06	1.81E+01
8	Containment bypass	5.54E-07	4.47%	8.34E+06	4.62E+00
	Total	1.24E-05			2.52E+01

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 in order to preserve total CDF.

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

The risk increase associated with extending the ILRT interval involves the potential that a core damage event that normally would result in only a small radioactive release from an intact containment could, in fact, result in a larger release due to the increase in probability of failure to detect a pre-existing leak. With strict adherence to the EPRI guidance, 100% of the Class 3b contribution would be considered LERF.

Regulatory Guide 1.174 [Reference 4] provides guidance for determining the risk impact of plant-specific changes to the licensing basis. RG 1.174 [Reference 4] defines "very small" changes in risk as resulting in increases of CDF less than 10^{-6}/year and increases in LERF less than 10^{-7}/year, and "small" changes in LERF as less than 10^{-6}/year. Since containment overpressure is not required in support of ECCS performance to mitigate design basis accidents and no equipment in the shield building is credited in the CDF model at BRW, the ILRT extension does not impact CDF. Therefore, the relevant risk-impact metric is LERF.

For BRW, 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 5-11 and Table 5-12, the Class 3b frequency is 8.95E-8/year for Unit 1 and 8.93E-8/year for Unit 2; based on a 15-year test interval from Table 5-13 and Table 5-14, the Class 3b frequency is 1.34E-7/year for Unit 1 and 1.34E-7/year for Unit 2. Thus, the increase in the overall probability of LERF due to Class 3b sequences that is due to increasing the ILRT test interval from 3 to 15 years is 1.07E-7/year for Unit 1 and 1.07E-7/year for Unit 2. Similarly, the increase due to increasing the interval from 10 to 15 years is 4.47E-8/year for Unit 1 and 4.46E-8/year for Unit 2. 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 meets the criteria for a "small" change when comparing the 15-year results to the original 3-year requirement. Table 5-15 summarizes these results.

Table 5-15 – Impact on LERF due to Extended Type A Testing Intervals

ILRT Inspection Interval	Unit 1: 3 Years (baseline)	Unit 1: 10 Years	Unit 1: 15 Years	Unit 2: 3 Years (baseline)	Unit 2: 10 Years	Unit 2: 15 Years
Class 3b (Type A LERF)	2.68E-08	8.95E-08	1.34E-07	2.68E-08	8.93E-08	1.34E-07
ΔLERF (3 year baseline)		6.26E-08	1.07E-07		6.25E-08	1.07E-07
ΔLERF (10 year baseline)			4.47E-08			4.46E-08

NEI 94-01 [Reference 1] 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. As shown in Table 5-16, the results of this calculation meet the dose rate criteria.

Table 5-16 – Impact on Dose Rate due to Extended Type A Testing Intervals

ILRT Inspection Interval	Unit 1: 10 Years	Unit 1: 15 Years	Unit 2: 10 Years	Unit 2: 15 Years
ΔDose Rate (3 year baseline)	2.100E-01	3.600E-01	2.095E-01	3.592E-01
ΔDose Rate (10 year baseline)		1.500E-01		1.496E-01
%ΔDose Rate (3 year baseline)	0.841%	1.442%	0.845%	1.448%
%ΔDose Rate (10 year baseline)		0.596%		0.598%

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.

The increase in the overall probability of LERF due to Class 3b sequences being slightly greater than 1E-7 is not unexpected. Since the RG 1.174 Region III threshold is exceeded, refinement is performed. Conservatism can be reduced by examining the source term expected to be available for release during the accident sequence. The source term is greatly reduced if the debris expelled from the reactor remains covered with water. Therefore, if the accident sequence contains containment spray success, the source term is not considered to lead to a large early release [Reference 35]. Excluding INTACT and LATE scenarios where containment spray is successful, and therefore scrubbing the source term release, results in a frequency reduction.

The exclusion of these frequencies leads to reduced risk being assigned to Class 3. The equations below show the Class 3b calculations for each ILRT interval, using the INTACT and LATE frequencies where containment spray is failed.

$$Freq_{U1class3b} = P_{class3b} * \left(INTACT_{CS_{fail}} + LATE_{CS_{fail}}\right) = \frac{.5}{217} * (0 + 4.40\text{E-}6) = 1.01\text{E-}8$$

$$Freq_{U2class3b} = P_{class3b} * \left(INTACT_{CS_{fail}} + LATE_{CS_{fail}}\right) = \frac{.5}{217} * (0 + 4.34\text{E-}6) = 9.95\text{E-}9$$

$$Freq_{U1Class3b10yr} = \frac{10}{3} * \frac{.5}{218} * \left(INTACT_{CS_{fail}} + LATE_{CS_{fail}}\right) = \frac{10}{3} * \frac{.5}{218} * 4.40\text{E-}6 = 3.37\text{E-}8$$

$$Freq_{U2Class3b10yr} = \frac{10}{3} * \frac{.5}{218} * \left(INTACT_{CS_{fail}} + LATE_{CS_{fail}}\right) = \frac{10}{3} * \frac{.5}{218} * 4.34\text{E-}6 = 3.32\text{E-}08$$

$$Freq_{U1Class3b15yr} = \frac{15}{3} * \frac{.5}{218} * \left(INTACT_{CS_{fail}} + LATE_{CS_{fail}}\right) = 5 * \frac{.5}{218} * 4.40\text{E-}6 = 5.05\text{E-}8$$

$$Freq_{U2Class3b15yr} = \frac{15}{3} * \frac{.5}{218} * \left(INTACT_{CS_{fail}} + LATE_{CS_{fail}}\right) = 5 * \frac{.5}{218} * 4.34\text{E-}6 = 4.97\text{E-}8$$

Substituting these values into the previously defined calculation method yields the final results displayed in Table 5-17.

Table 5-17 – Impact on LERF due to Extended Type A Testing Intervals – Containment Spray Credited

ILRT Inspection Interval	Unit 1: 3 Years (baseline)	Unit 1: 10 Years	Unit 1: 15 Years	Unit 2: 3 Years (baseline)	Unit 2: 10 Years	Unit 2: 15 Years
Class 3b (Type A LERF)	1.01E-08	3.37E-08	5.05E-08	9.95E-09	3.32E-08	4.97E-08
ΔLERF (3 year baseline)		2.36E-08	4.04E-08		2.32E-08	3.98E-08
ΔLERF (10 year baseline)			1.68E-08			1.66E-08

The adjusted containment spray inputs allow the Unit 1 and 2 values to be significantly less than the 1E-7 ΔLERF metric. The ΔLERF between the 3 years and the 15 years is 4.04E-8/yr for Unit 1 and 3.98E-8/yr for Unit 2. These values show the proposed extension meets the definition of a "very small" change in risk as defined in Regulatory Guide 1.174.

As shown in Table 5-18, the results of this calculation meet the dose rate criteria.

Table 5-18 – Impact on Dose Rate due to Extended Type A Testing Intervals – Containment Spray Credited

ILRT Inspection Interval	Unit 1: 10 Years	Unit 1: 15 Years	Unit 2: 10 Years	Unit 2: 15 Years
ΔDose Rate (3 year baseline)	7.901E-02	1.354E-01	7.779E-02	1.334E-01
ΔDose Rate (10 year baseline)		5.643E-02		5.557E-02
%ΔDose Rate (3 year baseline)	0.317%	0.544%	0.314%	0.539%
%ΔDose Rate (10 year baseline)		0.226%		0.224%

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.

Since CCFP is only concerned with a containment failure and not whether the release is small or large, the Class 1 results without containment spray refinement are used to calculate the CCFP. Table 5-19 shows the steps and results of this calculation.

Table 5-19 – Impact on CCFP due to Extended Type A Testing Intervals

ILRT Inspection Interval	Unit 1: 3 Years (baseline)	Unit 1: 10 Years	Unit 1: 15 Years	Unit 2: 3 Years (baseline)	Unit 2: 10 Years	Unit 2: 15 Years
f(ncf) (/yr)	6.71E-06	6.65E-06	6.60E-06	6.72E-06	6.66E-06	6.62E-06
f(ncf)/CDF	0.540	0.535	0.531	0.542	0.537	0.534
CCFP	0.460	0.465	0.469	0.458	0.463	0.466
ΔCCFP (3 year baseline)		0.504%	0.864%		0.504%	0.864%
ΔCCFP (10 year baseline)			0.360%			0.360%

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 0.864% for Unit 1 and 0.864% for Unit 2. 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 5-20, 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 BRW has a different containment type, this event could potentially occur at BRW (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 BRW.
- 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 5-20, 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 BRW, the ILRT maximum pressure is 42.8 psig [References 31 and 32]. 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 5-20, 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 5-20, 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 5-20, 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 5-20 – 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 1 average 5-10 15 15 year average = 6.44E-03	Failure rate 2.05E-03 5.19E-03 1.43E-02	Year 1 average 5-10 15 15 year average = 1.61E-03	Failure rate 5.13E-04 1.30E-03 3.57E-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 BRW.

Table 5-21 – Total Likelihood on Non-Detected Containment Leakage Due to Corrosion for BRW

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 Braidwood Fire PRA model was created to satisfy the ASME/ANS PRA standard [Reference 37] with guidance from NUREG/CR-6850 [Reference 29]. The BB016a2 Fire ASM model was used to obtain the fire CDF and LERF values [Reference 18]. When CAFTA calculates CDF and LERF, it uses the minimum-cutset-upper-bound (MCUB) approximation, which overestimates overall risk. The Advanced Cutset Upper Bound Estimator (ACUBE) improves the accuracy of the minimum cutsets calculation [Reference 39]. Many of the top CDF and LERF cutsets were processed with ACUBE to reduce the conservatism of the risk values calculated by CAFTA (a different number of cutsets were processed with ACUBE based on computational limitations for each case). To reduce conservatism in the model, the methodology of subtracting existing LERF from CDF is also applied to the Fire PRA model. The following shows the calculation for Class 3b:

$$Freq_{U1class3b} = P_{class3b} * (CDF - LERF) = \frac{0.5}{218} * (5.62E\text{-}5 - 5.39E\text{-}6) = 1.16\text{E-}7$$

$$Freq_{U1class3b10yr} = \frac{10}{3} * P_{class3b} * (CDF - LERF) = \frac{10}{3} * \frac{0.5}{218} * (5.62E\text{-}5 - 5.39E\text{-}6) = 3.88\text{E-}7$$

$$Freq_{U1class3b15yr} = \frac{15}{3} * P_{class3b} * (CDF - LERF) = 5 * \frac{0.5}{218} * (5.62E\text{-}5 - 5.39E\text{-}6) = 5.82\text{E-}7$$

$$Freq_{U2class3b} = P_{class3b} * (CDF - LERF) = \frac{0.5}{218} * (5.67E\text{-}5 - 5.18E\text{-}6) = 1.18\text{E-}7$$

$$Freq_{U2class3b10yr} = \frac{10}{3} * P_{class3b} * (CDF - LERF) = \frac{10}{3} * \frac{0.5}{218} * (5.67E\text{-}5 - 5.18E\text{-}6) = 3.94\text{E-}7$$

$$Freq_{U2class3b15yr} = \frac{15}{3} * P_{class3b} * (CDF - LERF) = 5 * \frac{0.5}{218} * (5.67E\text{-}5 - 5.18E\text{-}6) = 5.91\text{E-}7$$

Reference 28 was developed for risk-informed applications and provides an assessment of the seismic hazard. The Seismic CDF and LERF are estimated to be 4.2E-6/yr and 6.2E-7/yr, respectively (same for both units).

Subtracting seismic LERF from CDF, the Class 3b frequency can be calculated by the following formulas:

$$Freq_{class3b} = P_{class3b} * (CDF - LERF) = \frac{0.5}{218} * (4.20E\text{-}6 - 6.20E\text{-}7) = 8.21\text{E-}9$$

$$Freq_{class3b10yr} = \frac{10}{3} * P_{class3b} * (CDF - LERF) = \frac{10}{3} * \frac{0.5}{218} * (4.20E\text{-}6 - 6.20E\text{-}7) = 2.74\text{E-}8$$

$$Freq_{class3b15yr} = \frac{15}{3} * P_{class3b} * (CDF - LERF) = 5 * \frac{0.5}{218} * (4.20E\text{-}6 - 6.20E\text{-}7) = 4.11\text{E-}8$$

Reference 28 also provides a high winds and tornado assessment; the CDF associated with high winds and tornado missiles at Braidwood is estimated to be less than 1E-6/yr for both units. Since an exact CDF is not provided, a bounding value of 1E-6/yr is used for this analysis. Since no high winds and tornado LERF value is calculated, it is assumed the LERF/CDF ratio will be similar for high winds and tornado risk as for internal events risk. Applying the internal event LERF/CDF ratio to the high winds and tornado CDF estimate yields an estimated high winds and tornado LERF of 5.82E-8/yr for Unit 1 and 5.81E-8/yr for Unit 2, as shown by the equations below.

$$LERF_{U1HW\_T} \approx CDF_{U1HW\_T} * LERF_{U1IE} / CDF_{U1IE} = 1.00\text{E-}6 * 7.24\text{E-}7 / 1.24\text{E-}5 = 5.82\text{E-}8$$

$$LERF_{U2HW\_T} \approx CDF_{U2HW\_T} * LERF_{U2IE} / CDF_{U2IE} = 1.00\text{E-}6 * 7.21\text{E-}7 / 1.24\text{E-}5 = 5.81\text{E-}8$$

The methodology of subtracting existing LERF from CDF is also applied to high winds and tornado risk. The following shows the calculation for Class 3b:

$$Freq_{U1class3b} = P_{class3b} * (CDF - LERF) = \frac{0.5}{218} * (1.00E\text{-}6 - 5.82E\text{-}8) = 2.16\text{E-}9$$

$$Freq_{U1class3b10yr} = \frac{10}{3} * P_{class3b} * (CDF - LERF) = \frac{10}{3} * \frac{0.5}{218} * (1.00E\text{-}6 - 5.82E\text{-}8) = 7.20\text{E-}9$$

$$Freq_{U1class3b15yr} = \frac{15}{3} * P_{class3b} * (CDF - LERF) = 5 * \frac{0.5}{218} * (1.00E\text{-}6 - 5.82E\text{-}8) = 1.08\text{E-}8$$

$$Freq_{U2class3b} = P_{class3b} * (CDF - LERF) = \frac{0.5}{218} * (1.00E\text{-}6 - 5.81E\text{-}8) = 2.16\text{E-}9$$

$$Freq_{U2class3b10yr} = \frac{10}{3} * P_{class3b} * (CDF - LERF) = \frac{10}{3} * \frac{0.5}{218} * (1.00E\text{-}6 - 5.81E\text{-}8) = 7.20\text{E-}9$$

$$Freq_{U2class3b15yr} = \frac{15}{3} * P_{class3b} * (CDF - LERF) = 5 * \frac{0.5}{218} * (1.00E\text{-}6 - 5.81E\text{-}8) = 1.08\text{E-}8$$

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 5-22 for Unit 1 and Table 5-23 for Unit 2.

Table 5-22 – Unit 1 BRW External Event Impact on ILRT LERF Calculation

Hazard	EPRI Accident Class 3b Frequency			LERF Increase (from 3 per 10 years to 1 per 15 years)
	3 per 10 year	1 per 10 year	1 per 15 years	
External Events	1.27E-07	4.23E-07	6.34E-07	5.07E-07
Internal Events	1.01E-08	3.37E-08	5.05E-08	4.04E-08
Combined	1.37E-07	4.56E-07	6.84E-07	5.48E-07

Table 5-23 – Unit 2 BRW External Event Impact on ILRT LERF Calculation

Hazard	EPRI Accident Class 3b Frequency			LERF Increase (from 3 per 10 years to 1 per 15 years)
	3 per 10 year	1 per 10 year	1 per 15 years	
External Events	1.29E-07	4.29E-07	6.43E-07	5.14E-07
Internal Events	9.95E-09	3.32E-08	4.97E-08	3.98E-08
Combined	1.39E-07	4.62E-07	6.93E-07	5.54E-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 2.28E-7 for Unit 1 and 2.31E-7 for Unit 2; the total change in LERF due to increasing the ILRT interval from 3 to 15 years is 5.48E-7 for Unit 1 and 5.54E-7 for Unit 2, 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 is calculated below:

$$LERF = LERF_{internal} + LERF_{fire} + LERF_{seismic} + LERF_{HW} + LERF_{class3Bincrease}$$

$$LERF_{U1} = 7.24E\text{-}7/yr + 5.39E\text{-}6/yr + 6.20E\text{-}7/yr + 5.82E\text{-}8/yr + 5.48E\text{-}7/yr = 7.34E\text{-}6/yr$$

$$LERF_{U2} = 7.21E\text{-}7/yr + 5.18E\text{-}6/yr + 6.20E\text{-}7/yr + 5.81E\text{-}8/yr + 5.54E\text{-}7/yr = 7.13E\text{-}6/yr$$

As specified in Regulatory Guide 1.174 [Reference 4], since the total LERF is less than 1.0E-05, it is acceptable for the ΔLERF to be between 1.0E-07 and 1.0E-06.

5.2.7.1 Screened External Hazards

Several "other" external events were evaluated in the External Hazards Assessment [Reference 28]. As detailed in Section 2.2 of Reference 28, External Flooding is not a significant contributor to overall risk impact; therefore, it is screened from consideration. As detailed in Section 2.4 of Reference 28, many additional hazards were addressed in assessing the overall risk impact from External Hazards; all of these hazards were screened from consideration.

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. Preserve a reasonable balance among the layers of defense.

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. Preserve adequate capability of design features without an overreliance on programmatic activities as compensatory measures.

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

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. Preserve adequate defense against potential CCFs.

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. Maintain multiple fission product barriers.

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. Preserve sufficient defense against human errors.

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. Continue to meet the intent of the plant's design criteria.

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.

5.3 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 5-24 – Table 5-29. Since CCFP is only concerned with a containment failure and not whether the release is small or large, the Class 1 results without containment spray refinement is used to calculate the CCFP. The Annual Dose Rate calculations are performed using the containment spray adjustments. The steel liner corrosion likelihood was increased by a factor of 1000, 10000, and 100000. Except for extreme factors of 10000 and 100000, the corrosion likelihood is relatively insensitive to the results.

Table 5-24 – Unit 1 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	2.68E-08	8.95E-08	1.34E-07	6.26E-08	1.07E-07	4.48E-08
Corrosion Likelihood X 1000	2.71E-08	9.41E-08	1.50E-07	6.70E-08	1.23E-07	5.63E-08
Corrosion Likelihood X 10000	2.92E-08	1.36E-07	2.96E-07	1.07E-07	2.67E-07	1.61E-07
Corrosion Likelihood X 100000	5.07E-08	5.53E-07	1.75E-06	5.02E-07	1.70E-06	1.20E-06

Table 5-25 – Unit 2 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	2.68E-08	8.93E-08	1.34E-07	6.25E-08	1.07E-07	4.47E-08
Corrosion Likelihood X 1000	2.70E-08	9.39E-08	1.50E-07	6.69E-08	1.23E-07	5.62E-08
Corrosion Likelihood X 10000	2.92E-08	1.35E-07	2.96E-07	1.06E-07	2.66E-07	1.60E-07
Corrosion Likelihood X 100000	5.06E-08	5.51E-07	1.75E-06	5.01E-07	1.70E-06	1.20E-06

Table 5-26 – Unit 1 Steel Liner Corrosion Sensitivity: CCFP

	CCFP (3-per-10 year ILRT)	CCFP (1-per-10 year ILRT)	CCFP (1-per-15 year ILRT)	CCFP Increase (3-per-10 to 1-per-10)	CCFP Increase (3-per-10 to 1-per-15)	CCFP Increase (1-per-10 to 1-per-15)
Corrosion Likelihood X 1	4.60E-01	4.65E-01	4.69E-01	5.04E-03	8.64E-03	3.60E-03
Corrosion Likelihood X 1000	4.60E-01	4.65E-01	4.69E-01	5.08E-03	8.72E-03	3.63E-03
Corrosion Likelihood X 10000	4.60E-01	4.66E-01	4.70E-01	5.49E-03	9.41E-03	3.92E-03
Corrosion Likelihood X 100000	4.62E-01	4.72E-01	4.78E-01	9.52E-03	1.63E-02	6.80E-03

Table 5-27 – Unit 2 Steel Liner Corrosion Sensitivity: CCFP

	CCFP (3-per-10 year ILRT)	CCFP (1-per-10 year ILRT)	CCFP (1-per-15 year ILRT)	CCFP Increase (3-per-10 to 1-per-10)	CCFP Increase (3-per-10 to 1-per-15)	CCFP Increase (1-per-10 to 1-per-15)
Corrosion Likelihood X 1	4.58E-01	4.63E-01	4.66E-01	5.04E-03	8.64E-03	3.60E-03
Corrosion Likelihood X 1000	4.58E-01	4.63E-01	4.66E-01	5.09E-03	8.72E-03	3.63E-03
Corrosion Likelihood X 10000	4.58E-01	4.63E-01	4.67E-01	5.49E-03	9.41E-03	3.92E-03
Corrosion Likelihood X 100000	4.60E-01	4.69E-01	4.76E-01	9.52E-03	1.63E-02	6.80E-03

Table 5-28 – Unit 1 Steel Liner Corrosion Sensitivity: Dose Rate

	Dose Rate (3-per-10 year ILRT)	Dose Rate (1-per-10 year ILRT)	Dose Rate (1-per-15 year ILRT)	Dose Rate Increase (3-per-10 to 1-per-10)	Dose Rate Increase (3-per-10 to 1-per-15)	Dose Rate Increase (1-per-10 to 1-per-15)
Corrosion Likelihood X 1	9.00E-02	3.00E-01	4.50E-01	2.10E-01	3.60E-01	1.50E-01
Corrosion Likelihood X 1000	9.08E-02	3.03E-01	4.54E-01	2.12E-01	3.63E-01	1.51E-01
Corrosion Likelihood X 10000	9.80E-02	3.27E-01	4.90E-01	2.29E-01	3.92E-01	1.63E-01
Corrosion Likelihood X 100000	1.70E-01	5.67E-01	8.50E-01	3.97E-01	6.80E-01	2.83E-01

Table 5-29 – Unit 2 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	8.98E-02	2.99E-01	4.49E-01	2.10E-01	3.59E-01	1.50E-01
Corrosion Likelihood X 1000	9.06E-02	3.02E-01	4.53E-01	2.11E-01	3.62E-01	1.51E-01
Corrosion Likelihood X 10000	9.78E-02	3.26E-01	4.89E-01	2.28E-01	3.91E-01	1.63E-01
Corrosion Likelihood X 100000	1.70E-01	5.66E-01	8.48E-01	3.96E-01	6.79E-01	2.83E-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 5-30 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 5-30 – BRW 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 5-11 – Table 5-14 for the expert elicitation sensitivity yields the results in Table 5-31 for Unit 1 and Table 5-32 for Unit 2.

Table 5-31 – BRW Unit 1 Summary of ILRT Extension Using Expert Elicitation Values

Accident Class	ILRT Interval							
	3 per 10 Years				1 per 10 Years		1 per 15 Years	
	Base Frequency	Adjusted Base Frequency	Dose (person-rem)	Dose Rate (person-rem/yr)	Frequency	Dose Rate (person-rem/yr)	Frequency	Dose Rate (person-rem/yr)
1	6.74E-06	6.69E-06	2.48E+04	1.66E-01	6.58E-06	1.63E-01	6.50E-06	1.61E-01
2	5.78E-07	5.78E-07	3.25E+06	1.88E+00	5.78E-07	1.88E+00	5.78E-07	1.88E+00
3a	N/A	4.54E-08	2.48E+05	1.13E-02	1.51E-07	3.75E-02	2.27E-07	5.63E-02
3b	N/A	2.89E-09	2.48E+06	7.17E-03	9.64E-09	2.39E-02	1.45E-08	3.59E-02
7	4.56E-06	4.56E-06	3.99E+06	1.82E+01	4.56E-06	1.82E+01	4.56E-06	1.82E+01
8	5.54E-07	5.54E-07	8.34E+06	4.63E+00	5.54E-07	4.63E+00	5.54E-07	4.63E+00
Totals	1.24E-05	1.24E-05	1.83E+07	2.49E+01	1.24E-05	2.49E+01	1.24E-05	2.50E+01
ΔLERF (3 per 10 yrs base)	N/A				6.75E-09		1.16E-08	
ΔLERF (1 per 10 yrs base)	N/A				N/A		4.82E-09	
CCFP	45.82%				45.87%		45.91%	

Table 5-32 – BRW Unit 2 Summary of ILRT Extension Using Expert Elicitation Values

Accident Class	ILRT Interval							
	3 per 10 Years				1 per 10 Years		1 per 15 Years	
	Base Frequency	Adjusted Base Frequency	Dose (person-rem)	Dose Rate (person-rem/yr)	Frequency	Dose Rate (person-rem/yr)	Frequency	Dose Rate (person-rem/yr)
1	6.75E-06	6.70E-06	2.48E+04	1.66E-01	6.59E-06	1.63E-01	6.51E-06	1.61E-01
2	5.74E-07	5.74E-07	3.25E+06	1.86E+00	5.74E-07	1.86E+00	5.74E-07	1.86E+00
3a	N/A	4.53E-08	2.48E+05	1.12E-02	1.51E-07	3.75E-02	2.27E-07	5.62E-02
3b	N/A	2.88E-09	2.48E+06	7.15E-03	9.62E-09	2.38E-02	1.44E-08	3.58E-02
7	4.52E-06	4.52E-06	3.99E+06	1.81E+01	4.52E-06	1.81E+01	4.52E-06	1.81E+01
8	5.54E-07	5.54E-07	8.34E+06	4.62E+00	5.54E-07	4.62E+00	5.54E-07	4.62E+00
Totals	1.24E-05	1.24E-05	1.83E+07	2.47E+01	1.24E-05	2.48E+01	1.24E-05	2.48E+01
ΔLERF (3 per 10 yrs base)	N/A				6.73E-09		1.15E-08	
ΔLERF (1 per 10 yrs base)	N/A				N/A		4.81E-09	
CCFP	45.58%				45.63%		45.67%	

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

Table 6-1 – Unit 1 ILRT Extension Summary

Class	Dose (person-rem)	Base Case 3 in 10 Years		Extend to 1 in 10 Years		Extend to 1 in 15 Years	
		CDF/Year	Person-Rem/Year	CDF/Year	Person-Rem/Year	CDF/Year	Person-Rem/Year
1	2.48E+04	6.69E-06	1.66E-01	6.57E-06	2.48E+04	6.48E-06	1.61E-01
2	3.25E+06	5.78E-07	1.88E+00	5.78E-07	3.25E+06	5.78E-07	1.88E+00
3a	2.48E+05	4.06E-08	1.01E-02	1.35E-07	2.48E+05	2.03E-07	5.03E-02
3b	2.48E+06	1.01E-08	2.51E-02	3.37E-08	2.48E+06	5.05E-08	1.25E-01
7	3.99E+06	4.56E-06	1.82E+01	4.56E-06	3.99E+06	4.56E-06	1.82E+01
8	8.34E+06	5.54E-07	4.63E+00	5.54E-07	8.34E+06	5.54E-07	4.63E+00
Total		**1.24E-05**	**2.49E+01**	**1.24E-05**	**2.50E+01**	**1.24E-05**	**2.51E+01**

ILRT Dose Rate from 3a and 3b

		Base Case 3 in 10 Years	Extend to 1 in 10 Years	Extend to 1 in 15 Years
ΔTotal Dose Rate	From 3 Years	N/A	7.901E-02	1.354E-01
	From 10 Years	N/A	N/A	5.643E-02
%ΔDose Rate	From 3 Years	N/A	0.317%	0.544%
	From 10 Years	N/A	N/A	0.226%

3b Frequency (LERF)

		Base Case 3 in 10 Years	Extend to 1 in 10 Years	Extend to 1 in 15 Years
ΔLERF	From 3 Years	N/A	2.36E-08	4.04E-08
	From 10 Years	N/A	N/A	1.68E-08

CCFP %

		Base Case 3 in 10 Years	Extend to 1 in 10 Years	Extend to 1 in 15 Years
ΔCCFP%	From 3 Years	N/A	0.504%	0.864%
	From 10 Years	N/A	N/A	0.360%

Table 6-2 – Unit 2 ILRT Extension Summary

Class	Dose (person-rem)	Base Case 3 in 10 Years		Extend to 1 in 10 Years		Extend to 1 in 15 Years	
		CDF/Year	Person-Rem/Year	CDF/Year	Person-Rem/Year	CDF/Year	Person-Rem/Year
1	2.48E+04	6.70E-06	1.66E-01	6.58E-06	1.63E-01	6.50E-06	1.61E-01
2	3.25E+06	5.74E-07	1.86E+00	5.74E-07	1.86E+00	5.74E-07	1.86E+00
3a	2.48E+05	4.00E-08	9.91E-03	1.33E-07	3.30E-02	2.00E-07	4.96E-02
3b	2.48E+06	9.95E-09	2.47E-02	3.32E-08	8.22E-02	4.97E-08	1.23E-01
7	3.99E+06	4.52E-06	1.81E+01	4.52E-06	1.81E+01	4.52E-06	1.81E+01
8	8.34E+06	5.54E-07	4.62E+00	5.54E-07	4.62E+00	5.54E-07	4.62E+00
Total		**1.24E-05**	**2.47E+01**	**1.24E-05**	**2.48E+01**	**1.24E-05**	**2.49E+01**
ILRT Dose Rate from 3a and 3b							
ΔTotal Dose Rate	From 3 Years	N/A		7.779E-02		1.334E-01	
	From 10 Years	N/A		N/A		5.557E-02	
%ΔDose Rate	From 3 Years	N/A		0.314%		0.539%	
	From 10 Years	N/A		N/A		0.224%	
3b Frequency (LERF)							
ΔLERF	From 3 Years	N/A		2.32E-08		3.98E-08	
	From 10 Years	N/A		N/A		1.66E-08	
CCFP %							
ΔCCFP%	From 3 Years	N/A		0.504%		0.864%	
	From 10 Years	N/A		N/A		0.360%	

7.0 CONCLUSIONS AND RECOMMENDATIONS

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

- Regulatory Guide 1.174 [Reference 4] provides guidance for determining the risk impact of plant-specific changes to the licensing basis. Regulatory Guide 1.174 defines "very small" changes in risk as resulting in increases of CDF less than 1.0E-06/year and increases in LERF less than 1.0E-07/year. Since the ILRT does not impact CDF, the relevant criterion is LERF. The increase in LERF resulting from a change in the Type A ILRT test interval from 3 in 10 years to 1 in 15 years is estimated as 4.04E-8/year for Unit 1 and 3.98E-8/year for Unit 2 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 1.68E-8/year for Unit 1 and 1.66E-8/year for Unit 2, 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 5.48E-7/year for Unit 1 and 5.54E-7/year for Unit 2 using the EPRI guidance, and total LERF is 7.34E-6/year for Unit 1 and 7.13E-6/year for Unit 2. 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 2.28E-7/year for Unit 1 and 2.31E-7/year for Unit 2, and the total LERF is 7.03E-6 /year for Unit 1 and 6.81E-6/year for Unit 2. Therefore, the risk increase is "small" using the acceptance guidelines of Regulatory Guide 1.174 [Reference 4]. As discussed in Sections 5.1.3 and 5.2.7, the EPRI methodology used to estimate the increase in LERF and the models used to estimate total LERF are conservative. Therefore, even though the total LERF is near the Regulatory Guide 1.174 threshold, the conservative methodology adds margin.
- 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 0.135 person-rem/year for Unit 1 and 0.133 person-rem/year for Unit 2. NEI 94-01 [Reference 1] 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 0.864% for Unit 1 and 0.864% for Unit 2. 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 BRW 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 BRW confirm these general conclusions on a plant-specific basis considering the severe accidents evaluated for BRW, the BRW containment failure modes, and the local population surrounding BRW.

A. PRA ACCEPTABILITY

A.1. PRA Quality Statement for Permanent 15-Year ILRT Extension

The BB016a2 version of the Braidwood PRA model is the most recent evaluation of internal event risk. The Braidwood PRA modeling is highly detailed, including a wide variety of initiating events, modeled systems, operator actions, and common cause events. The PRA model quantification process used for the Braidwood PRA is based on the single top fault tree methodology, which is a well-known PRA methodology in the industry. The Braidwood internal events and Fire PRA models have been peer reviewed to RG 1.200 Rev 2. The internal events model and Fire PRA have open F&Os; further details are provided section A.2.

Exelon Generation Company, LLC (Exelon) employs a multi-faceted approach to establishing and maintaining the technical adequacy and plant fidelity of the PRA models for all operating Exelon nuclear generation sites. This approach includes both a proceduralized PRA maintenance and update process and the use of self-assessments and independent peer reviews. The following information describes the Exelon approach to PRA model maintenance, as it applies to the Braidwood PRA.

A.1.1 PRA Maintenance and Update

The Exelon risk management process ensures that the applicable PRA model is an accurate reflection of the as-built and as-operated plants. This process is defined in the Exelon Risk Management program, which consists of a governing procedure (ER-AA-600, "Risk Management") and subordinate implementation training and reference materials (T&RMs).

- Exelon T&RM ER-AA-600-1015, "Full Power Internal Event (FPIE) PRA Model Update," delineates the responsibilities and guidelines for updating the full power internal events PRA models at all operating Exelon nuclear generation sites.
- ER-AA-600-1061, "Fire PRA Model Update and Control," delineates the responsibilities and guidelines for updating the station fire PRA.

The overall Exelon Risk Management program, including ER-AA-600-1015 and ER-AA-6001061, defines the process for implementing regularly scheduled and interim PRA model updates, tracking issues identified as potentially affecting the PRA models (e.g., due to changes in the plant, industry operating experience, etc.), and controlling the model and associated computer files. To ensure the current PRA model remains an accurate reflection of the as-built, as-operated plants, the following activities are routinely performed:

- Documentation of the PRA model, PRA products, and bases documents.
- The approach for controlling electronic storage of Risk Management (RM) products including PRA update information, PRA models, and PRA applications.
- Guidelines for updating the full power, internal events PRA models for Exelon nuclear generation sites.
- Guidance for use of quantitative and qualitative risk models in support of the On-Line Work Control Process Program for risk evaluations for maintenance tasks (corrective maintenance, preventive maintenance, minor maintenance, surveillance tests and modifications) on systems, structures, and components (SSCs) within the scope of the Maintenance Rule (10 CFR 50.65(a)(4)).

As indicated previously, RG 1.200 also requires that additional information be provided as part of the LAR submittal to demonstrate the technical adequacy of the PRA model used for the risk assessment. Each of these items (plant changes not yet incorporated into the PRA model, relevant peer review findings, and consistency with applicable PRA Standards) is discussed.

A.1.2 Plant Changes Not Yet Incorporated into the PRA Model

Each Exelon station maintains an updating requirements evaluation (URE) database to track all enhancements, corrections, and unincorporated plant changes. During the normal screening conducted as part of the plant change process, if a potential model update is identified, a new URE database item is created. Depending on the potential impact of the identified change, the requirements for incorporation will vary.

As part of this PRA evaluation, a review of open items in the URE database for Braidwood is performed, and an assessment of the impact on the results of the application is made. A few open UREs may lead to insignificant changes in CDF and LERF. Therefore, the model is adequate to perform this ILRT extension analysis.

A.2. Applicability of Peer Review Facts and Observations (F&Os)

The technical acceptability of the Braidwood PRA models has been demonstrated by the peer review process. The purpose of the industry PRA peer review process is to provide a method for establishing the technical capability and adequacy of a PRA relative to expectations of knowledgeable practitioners, using a set of guidance that establishes a set of minimum requirements. PRA peer reviews continue to be performed as PRAs are updated (and upgraded) to ensure the ability to support risk-informed applications and has proven to be a valuable process for establishing technical adequacy of nuclear power plant PRAs.

The internal events PRA model was subject to a self-assessment and a full-scope peer review conducted in July 2013. The fire PRA model was subject to a self-assessment and a full-scope peer review conducted in October 2015. Findings were reviewed and closed using the process documented in Reference 40 as accepted by NRC [Reference 41]. The closure review was conducted in February 2017 [Reference 33]. The F&O Closure Review consisting of an assessment of existing finding-level F&Os for the full-power internal events probabilistic risk assessment (FPIE PRA), the internal flood PRA (IFPRA), and the internal fire PRA (FPRA) was performed for Braidwood Units 1 and 2. Table A-1 provides a summary of open findings and dispositions of the Braidwood F&O Closure Review. This information demonstrates the PRA is of sufficient quality and level of detail to support the ILRT extension analysis.

Table A-1– Disposition and Resolution of Open Peer Review Findings and Self-Assessment Open Items

Finding Number	Supporting Requirement(s)	Capability Category (CC)	Finding Description	Disposition for ILRT Extension
SY-12-01	SY-B12	Cat I/II/III	Recovery actions are included in the model. However, a heating, ventilation and air conditioning (HVAC) dependency is not included for the switchgear and battery rooms. The room heatup calculations are based on a 4 hour mission time, with assumption that within that period, operators would take action to restore cooling to the rooms. A similar issue also was identified as part of the Supporting Requirement (SR) H2-E2 review where local operator actions to provide inverter room ventilation cooling (such as opening the door and installing temporary fans) were not identified. This was identified as a necessary operator action in the Equipment Survivability Notebook to recover from a failed room ventilation cooling system.	PARTIALLY RESOLVED Updated room cooling calculations and survivable temperature evaluations have been performed and indicate support for the current modeling assumptions for most scenarios for the Engineered Safety Feature (ESF) and Non-ESF Switchgear Rooms. The only identified potential impact on the model is for high energy line break (HELB) scenarios, which are an overall small contributor to the baseline risk results. A sensitivity calculation that simply inserts new HELB-related room cooling requirements shows a small increase in baseline CDF and LERF, but these increases do not trigger consideration of an emergent model update per the Exelon Risk Management procedures. Model changes to incorporate these additional HELB-related room cooling requirements are being tracked in the URE (Updating Requirements Evaluation) database and will be incorporated as part of the normal cumulative model update process. This open item will have minimal impact on the PRA results and minimal impact on this application. There is no impact to the FPRA from the outstanding HELB issue.
16-4	CS-B1	Cat I	Based on information provided there is lack of details to meet Capability Category II/III as the only statement is in Section 3.8 of the 'Braidwood Fire PRA Cable Selection Notebook (BW-PRA-021.03), Rev 0' : "The Braidwood Fire PRA reviewed the electrical coordination calculations for applicability to the Fire PRA. These were reviewed for each of the credited power supplies in the model." This did not provide analysis or identify any additional requirements; it only stated that it was reviewed.	This finding was resolved by reviewing the circuit breaker coordination calculations. Section 3.1.31 of Reference 18 provides a summary of the resolution. Attachment 5 of Reference 18 details the changes made on the 16a2 model. Since the model used for the ILRT extension analysis included this resolution, there is no impact to the ILRT extension analysis.

Table A-1– Disposition and Resolution of Open Peer Review Findings and Self-Assessment Open Items

Finding Number	Supporting Requirement(s)	Capability Category (CC)	Finding Description	Disposition for ILRT Extension
18-12	UNC-A1	Not Met	Some anomalies were observed in the database that was used for the uncertainty analysis which was different from that used in the FPRA.	This was resolved by reviewing and updating the current .rr files used for the uncertainty analysis with the correct parametric uncertainties. Since the model used for the ILRT analysis included this resolution, there is no impact to the ILRT extension analysis.
19-8	FQ-E1	Not Met	Document the relative contribution of contributors to LERF.	Importance reports by component are documented in Attachment 6 of Reference 18. Since this has been resolved and only involved documentation, there is no impact to the ILRT extension analysis.
19-9	FQ-E1	Not Met	HLR-QU-D7 requires review of importance of components and basic events to determine that they make logical sense.	Importance reports by component are documented in Attachment 6 of Reference 18. Since this has been resolved and only involved documentation, there is no impact to the ILRT extension analysis.
19-11	FQ-F1	Not Met	There is no documentation of the importance measures for Braidwood Unit 2 CDF/LERF from QU-F2 (j).	Importance reports by component are documented in Attachment 6 of Reference 18. Since this has been resolved and only involved documentation, there is no impact to the ILRT extension analysis.
19-15	FQ-F1	Not Met	Document the process used to identify plant damage states and accident progression contributors.	Plant Damage State to CDF Sequence mapping and plant damage state risk contributions are provided in Section 3.1.9 of Reference 18. Since this has been resolved and only involved documentation, there is no impact to the ILRT extension analysis.

Table A-1– Disposition and Resolution of Open Peer Review Findings and Self-Assessment Open Items

Finding Number	Supporting Requirement(s)	Capability Category (CC)	Finding Description	Disposition for ILRT Extension
20-1	IGN-A7	Cat I/II/III	During the Peer Review walkdown, three rooms were checked and found to have errors in ignition source counting: 11.4A-1, 11.4A-2, and 11.6-0.	Section 3.1.30 of Reference 18 summarizes the resolution. This was resolved by reviewing drawings to confirm where the four panels in question were located. After the review it was determined panels 1PL85JA/B are in 11.4A-1; 2PL85JA is in 11.4-0; and, 2PL85JB is in 11.4A-2. The FPRA 16a2 model has been updated to reflect these changes. Attachment 1 of Reference 18 provides the relevant scenario definition reports. Since the model used for the ILRT extension analysis included this resolution, there is no impact to the ILRT extension analysis.
20-8	FSS-B2	Cat I	Currently there is no credit given for operators safely shutting down the plant from the remote shutdown panel or from actions outside the MCR. This results in a CCDP of 1.0 being applied to scenarios that require operator abandonment or where sufficient functionality is lost at the Main Control Board (MCB).	Section 3.1.15 of Reference 18 summarizes the resolution. The FPRA is updated to credit operators safely shutting down the plant from the remote shutdown panel or from actions outside the MCR. The baseline control room abandonment conditional core damage probability (CCDP) is calculated using a qualitative assessment of the significance of the plant threat caused by the postulated fire event. This qualitative assessment uses the calculated CCDP associated with the fire impacts associated with the postulated source fire event. The intent of the criteria is to ensure that the CCDP is an appropriate bounding value. Since the model used for the ILRT extension analysis included this resolution, there is negligible impact to the ILRT extension analysis.

Table A-1– Disposition and Resolution of Open Peer Review Findings and Self-Assessment Open Items

Finding Number	Supporting Requirement(s)	Capability Category (CC)	Finding Description	Disposition for ILRT Extension
24-12	HRA-D1	Not Met	The joint human error probabilities (JHEPs) used to recover the risk are not supported by the documented dependency analysis.	PARTIALLY RESOLVED WITH OPEN DOCUMENTATION This finding remains open as a tracking item to provide a means of confirmation of consistency between the final FPRA human reliability analysis (HRA) report and the recovery file used in the final FPRA quantification model. Several changes were made during the final FPRA quantification and the final documentation will reflect these. However, a review of the final data to ensure that no inconsistencies exist is the intent of this open item. This is a documentation issue with no impact on this application.
24-14	UNC-A1	Not Met	A large number of entries in the uncertainty database lack sufficient information for a complete parametric uncertainty analysis.	This was resolved by reviewing and updating the current .rr files used for the uncertainty analysis with the correct parametric uncertainties. Since the model used for the ILRT analysis included this resolution, there is no impact to the ILRT extension analysis.
25-5	FQ-E1	Not Met	Some of the top scenarios are found as being not fully developed which may mask the important contributors to fire risk. This Supporting Requirement (SR) requires that significant contributors be identified in accordance with HLR-QU-D. HLR-QU-D6 requires that significant contributors be identified and HLR-QU-D7 requires review of important components and basic events (Bes) to determine that they make logical sense. This is not possible with overly conservative scenario models.	Importance reports by component are documented in Attachment 6 of Reference 18. As discussed in Section 3.2 or Reference 18, refinements were performed to address conservative joint human error probability (JHEP) values and to refine the treatment for the containment isolation valves for the containment mini-purge lines. These changes reduced the overall risk. Since the model used for the ILRT analysis included this resolution, there is no impact to the ILRT extension analysis.

Table A-1– Disposition and Resolution of Open Peer Review Findings and Self-Assessment Open Items

Finding Number	Supporting Requirement(s)	Capability Category (CC)	Finding Description	Disposition for ILRT Extension
25-9	FQ-E1	Not Met	HLR-QU-D7 requires review of importance of components and basic events (Bes) to determine that they make logical sense. The information provided did not meet the intent of providing a review of importance of components and basic events.	Importance reports by component are documented in Attachment 6 of Reference 18. Since this has been resolved and only involved documentation, there is no impact to the ILRT extension analysis.
25-11	PRM-B2	Not Met	Internal events finding AS-B3-01 does not appear to have been addressed for the FPRA analysis.	This finding was addressed via incorporation of the BB016a2 Full Power Internal Events / Flooding model which uses the new sump clogging value [18]. Since the model used for the ILRT analysis included this resolution, there is no impact to the ILRT extension analysis.
25-21	FQ-F1	Not Met	The documentation for the relative contribution of contributors to LERF was not addressed.	Importance reports by component are documented in Attachment 6 of Reference 18. Since this has been resolved and only involved documentation, there is no impact to the ILRT extension analysis.
25-22	FQ-F1	Not Met	The process used to identify plant damage states and accident progression contributors was not documented.	Plant Damage State to CDF Sequence mapping and plant damage state risk contributions are provided in Section 3.1.9 of Reference 18. Since this has been resolved and only involved documentation, there is no impact to the ILRT extension analysis.
26-9	IGN-A7	Cat I/II/III	During the Peer Review walkdown, two fairly large electrical wall-mounted cabinets with 14 switches were not counted in 11.4C-0, which is a risk-significant fire zone.	The recommendation for this finding was to establish and document the criteria for counting wall-mounted cabinets consistent with NRC guidance. This was resolved by performing walkdowns for wall-mounted cabinets throughout the plant. Data from the walkdowns were incorporated into the 16a2 model. Panel 0CO24E was added to PAU 18.11-0 and scenario 18.11-0_BASE_G [18]. Since the model used for the ILRT analysis included this resolution, there is no impact to the ILRT extension analysis.