Attachment 5

Evaluation of Risk Significance of Permanent ILRT Extension

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Advancing the Science of Safety

McGuire Nuclear Station:
Evaluation of Risk Significance of Permanent ILRT Extension

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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) to fifteen years. The extension would allow for substantial cost savings as the ILRT could be deferred for additional scheduled refueling outages for the McGuire Nuclear Station (MNS). The risk assessment follows the guidelines from NEI 94-01, Revision 3-A [Reference 1], the methodology used in EPRI TR-104285 [Reference 2], the NEI "Interim Guidance for Performing Risk Impact Assessments in Support of One-Time Extensions for Containment Integrated Leakage Rate Test Surveillance Intervals" from November 2001 [Reference 3], the NRC regulatory guidance on the use of Probabilistic Risk Assessment (PRA) as stated in Regulatory Guide 1.200 as applied to ILRT interval extensions, 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 McGuire 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 10CFR50, Appendix J (Option B) allow individual plants to extend the Integrated Leak Rate Test (ILRT) Type A surveillance testing frequency requirement from three in ten years to at least once in ten years. The revised Type A frequency is based on an acceptable performance history defined as two consecutive periodic Type A tests at least 24 months apart in which the calculated performance leakage rate was less than limiting containment leakage rate of $1L_a$.

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 assessment of the risk impact (in terms of increased public dose) associated with a range of extended leakage rate test intervals. To supplement the NRC's rulemaking basis, NEI undertook a similar study. The results of that study are documented in Electric Power Research Institute (EPRI) Research Project TR-104285, "Risk Impact Assessment of Revised Containment Leak Rate Testing Intervals."

The NRC report on performance-based leak testing, NUREG-1493, analyzed the effects of containment leakage on the health and safety of the public and the benefits realized from the containment leak rate testing. In that analysis, it was determined that for a representative PWR plant (i.e., Surry), that containment isolation failures contribute less than 0.1% 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 MNS.

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 Evaluation Reports (SER) 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 very small population dose is defined as an increase from the baseline interval (3 tests per 10 years) dose of ≤1.0 person-rem per year or 1% of the total baseline dose, whichever is less restrictive for the risk impact assessment of the proposed extended ILRT interval.

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

3.0 REFERENCES

The following references were used in this calculation:

1. *Revision 3-A to Industry Guideline for Implementing Performance-Based Option of 10 CFR Part 50, Appendix J*, NEI 94-01, July 2012.
2. *Risk Impact Assessment of Revised Containment Leak Rate Testing Intervals*, EPRI, Palo Alto, CA EPRI TR-104285, August 1994.
3. *Interim Guidance for Performing Risk Impact Assessments in Support of One-Time Extensions for Containment Integrated Leakage Rate Test Surveillance Intervals*, Revision 4, developed for NEI by EPRI and Data Systems and Solutions, November 2001.
4. An Approach for Using Probabilistic Risk Assessment in Risk-Informed Decisions on Plant-Specific Changes to the Licensing Basis, Regulatory Guide 1.174, May 2011.
5. *Response to Request for Additional Information Concerning the License Amendment Request for a One-Time Integrated Leakage Rate Test Extension*, Letter from Mr. C. H. Cruse (Calvert Cliffs Nuclear Power Plant) to NRC Document Control Desk, Docket No. 50-317, March 27, 2002.
6. Performance-Based Containment Leak-Test Program, NUREG-1493, September 1995.
7. *Evaluation of Severe Accident Risks: Surry Unit 1*, Main Report NUREG/CR-4551, SAND86-1309, Volume 3, Revision 1, Part 1, October 1990.
8. Letter from R. J. Barrett (Entergy) to U. S. Nuclear Regulatory Commission, IPN-01-007, January 18, 2001.
9. United States Nuclear Regulatory Commission, Indian Point Nuclear Generating Unit No. 3 – Issuance of Amendment Re: Frequency of Performance-Based Leakage Rate Testing (TAC No. MB0178), April 17, 2001.
10. *Impact of Containment Building Leakage on LWR Accident Risk*, Oak Ridge National Laboratory, NUREG/CR-3539, ORNL/TM-8964, April 1984.
11. *Reliability Analysis of Containment Isolation Systems*, Pacific Northwest Laboratory, NUREG/CR-4220, PNL-5432, June 1985.
12. Technical Findings and Regulatory Analysis for Generic Safety Issue II.E.4.3 'Containment Integrity Check', NUREG-1273, April 1988.
13. *Review of Light Water Reactor Regulatory Requirements*, Pacific Northwest Laboratory, NUREG/CR-4330, PNL-5809, Volume 2, June 1986.
14. Shutdown Risk Impact Assessment for Extended Containment Leakage Testing Intervals Utilizing ORAM™, EPRI, Palo Alto, CA, TR-105189, Final Report, May 1995.
15. *Severe Accident Risks: An Assessment for Five U. S. Nuclear Power Plants*, NUREG-1150, December 1990.
16. United States Nuclear Regulatory Commission, Reactor Safety Study, WASH-1400, October 1975.
17. Calculation MCC-1535.00-00-0183, Revision 5, McGuire Nuclear Station, "McGuire Rev. 4a PRA Model Integration."
18. McGuire Nuclear Station Fire PRA, MR3a Revision 2 version 24.
19. Calculation MCC-1535.07-00-0020, "Risk Assessment of MNS Integrated Leak Rate Test Extension," McGuire Units 1 and 2, Revision 4, February 2007.

20. Anthony R. Pietrangelo, One-time extensions of containment integrated leak rate test interval – additional information, NEI letter to Administrative Points of Contact, November 30, 2001.

21. Letter from J. A. Hutton (Exelon, Peach Bottom) to U. S. Nuclear Regulatory Commission, Docket No. 50-278, License No. DPR-56, LAR-01-00430, dated May 30, 2001.

22. *Risk Assessment for Joseph M. Farley Nuclear Plant Regarding ILRT (Type A) Extension Request*, prepared for Southern Nuclear Operating Co. by ERIN Engineering and Research, P0293010002-1929-030602, March 2002.

23. Letter from D. E. Young (Florida Power, Crystal River) to U. S. Nuclear Regulatory Commission, 3F0401-11, dated April 25, 2001.

24. *Risk Impact Assessment of Extended Integrated Leak Rate Testing Intervals*, Revision 2-A of 1009325, EPRI, Palo Alto, CA. 1018243, October 2008.

25. Risk Assessment for Vogtle Electric Generating Plant Regarding the ILRT (Type A) Extension Request, prepared for Southern Nuclear Operating Co. by ERIN Engineering and Research, February 2003.

26. Perspectives Gained from the IPEEE Program, USNRC, NUREG-1742, April 2002.

27. Procedure PT/2/A/4200/001, Revision 18, McGuire Nuclear Station, "Containment Integrated Leak Rate Test."

28. Letter L-14 -121, ML14111A291, FENOC Evaluation of the Proposed Amendment, Beaver Valley Power Station, Unit Nos. 1 and 2, April 2014.

29. Technical Letter Report ML112070867, Containment Liner Corrosion Operating Experience Summary, Revision 1, August 2011.

30. ML021580235, Duke Energy Corporation, "One-Time Extension of Integrated Leak Rate Testing (ILRT) Interval," May 29, 2002.

31. Armstrong, J., Simplified Level 2 Modeling Guidelines: WOG PROJECT: PA-RMSC-0088, Westinghouse, WCAP-16341-P, November 2005.

32. Generic Issue 199 (GI-199), ML100270582, September 2010, "Implications of Updated Probabilistic Seismic Hazard Estimates in Central and Eastern United States on Existing Plants: Safety/Risk Assessment."

33. An Approach for Estimating the Frequencies of Various Containment Failure Modes and Bypass Events, Brookhaven National Laboratory, NUREG/CR-6595, BNL-NUREG-52539, Revision 1.

34. Calculation MCC-1535.00-00-0004, "Seismic PRA/IPEEE Backup Calculations," McGuire Nuclear Station Units 1 & 2, Revision 14, August 1995.

35. ML14083A586, EPRI Evaluation, "Fleet Seismic Core Damage Frequency Estimates for Central and Eastern U.S. Nuclear Power Plants Using New Site-Specific Seismic Hazard Estimates," March 11, 2014.

36. Calculation MCC-1535.00-00-0178, "MNS High Wind Probabilistic Risk Assessment (HWPRA)," Revision 2, August 2016.

37. Calculation MCC-1535.00-00-0204, "McGuire Nuclear Station PRA Peer Review F&O Resolutions," Revision 0.

38. Calculation MCC-1535.00-00-0156, "McGuire LERF Methodology," Revision 2, March 2016.

39. ML13276A127, Duke Energy Carolinas, McGuire Nuclear Station, Transition to 10 CFR 50.48(c) - NFPA 805 Performance-Based Standard for Fire Protection for Light Water Reactor Electric Generating Plants, 2001 Edition, Transition Report, September 2013.

40. Duke letter to the NRC, ML15244B319, August 20, 2015.

41. "The Nuclear Energy Institute - Seismic Risk Evaluations for Plants in the Central and Eastern United States," ML14083A596, March 2014.

42. ML030210432, Duke Energy Corporation, "One-Time Extension of Integrated Leak Rate Testing (ILRT) Interval," January 8, 2003.

43. Calculation MCC-1535.00-00-0207, "Documentation of MR4 Model Update for Internal Flooding," Revision 1.

4.0 ASSUMPTIONS AND LIMITATIONS

The following assumptions were used in the calculation:

- The technical adequacy of the MNS 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 Attachment 1.
- The MNS Level 1 and LERF internal events PRA models provide representative results. The current internal events PRA model (Revision 4a) does not contain a full Level 2 PRA, but previous models contain a full Level 2 PRA. Where detail is needed from a Level 2 PRA, the results from the previous revisions are scaled using the current revision's total risk. It is a reasonable assumption that this scaling does not significantly affect the conclusions of this analysis.
- Even though MNS has two units, there is only one internal events PRA model because the two units are very similar. It is assumed that the two units are similar enough that the one internal events PRA model accurately models both units.
- It is appropriate to use the MNS internal events PRA model to effectively describe the risk change attributable to the ILRT extension. An extensive sensitivity study is done in Section 5.3.1 to show the effect of including external event models for the ILRT extension. The seismic risk from GI-199 [Reference 32] and the detailed Fire PRA (model fire-mr3a_r2v24) [Reference 18] are used for this sensitivity analysis.
- Accident classes describing radionuclide release end states are defined consistent with EPRI methodology [Reference 2].
- 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.

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 MNS. 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 0.04 person-rem per year..."

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

NUREG-1150 and the technical basis, NUREG/CR-4551, provide an ex-plant consequence analysis for a spectrum of accidents including a severe accident with the containment remaining intact (i.e., Tech Spec Leakage). This ex-plant consequence analysis is calculated for the 50-mile radial area surrounding Surry. The ex-plant calculation can be delineated to total person-rem for each identified Accident Progression Bin (APB) from NUREG/CR-4551. With the MNS Level 2 model end-states assigned to one of the NUREG/CR-4551 APBs, it is considered adequate to represent MNS. (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 MNS 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 MNS ILRT Extension Risk Assessment includes the following:

- CDF and LERF Model results [Reference 17]
- Release category definitions used in the Level 2 Model [Reference 19]
- Dose within a 50-mile radius [Reference 30]
- ILRT results to demonstrate adequacy of the administrative and hardware issues [Reference 30]

MNS Model

The Internal Events PRA Model that is used for MNS is characteristic of the as-built plant. The current Level 1 model (MNS PRA Model Version mr4a) [Reference 17] is a linked fault tree model. The total CDF is 2.38E-5/year; the total LERF is 1.85E-6 [Table 6 of Reference 17]. This model includes high winds risk. When high winds risk is excluded, Internal Events CDF is 1.34E-5, and Internal Events LERF is 9.12E-7. Table 5-1 and Table 5-2 provide a summary of the Internal Events CDF and LERF results for MNS PRA Model Version mr4a.

Unit 1 Fire CDF is 3.45E-5/year, Unit 1 Fire LERF is 4.30E-6/year, Unit 2 Fire CDF is 4.17E-5/year, and Unit 2 Fire LERF is 5.04E-6/year [Reference 18]. High Winds CDF is 1.03E-5, and High Winds LERF is 9.40E-7 [Table 6 of Reference 17]. The seismic risk is taken from GI-199 [Reference 32]. Refer to Section 5.3.1 for further details on external events as they pertain to this analysis.

Table 5-1 – Internal Events CDF (MNS PRA Model Version mr4a)

Internal Events	Frequency (per year)
Internal Floods	9.20E-06
Transients	2.85E-06
LOCAs	3.27E-07
SGTR	3.84E-07
RPV Rupture	2.90E-08
ISLOCA	6.34E-09
One or more Pressurizer Safety Valves Fail to Reseat	6.42E-07
Total Internal Events CDF	1.34E-05

Table 5-2 – Internal Events LERF (MNS PRA Model Version mr4a)

Internal Events	Frequency (per year)
Internal Floods	4.03E-07
SGTR	3.89E-07
Transients	1.03E-07
ISLOCA	6.34E-09
LOCAs	3.43E-09
RPV Rupture	6.39E-10

Table 5-2 – Internal Events LERF (MNS PRA Model Version mr4a)

Internal Events	Frequency (per year)
One or more Pressurizer Safety Valves Fail to Reseat	7.42E-09
Total Internal Events LERF	9.12E-07

Population Dose Calculations

The population dose calculation was reported in the "Risk Assessment of MNS Integrated Leak Rate Test Extension" [Reference 19]. Table 5-3 presents dose exposures calculated from methodology described in Reference 1 and data from Reference 19. Reference 19 provides four separate containment end-states that lead to Class 7; the Class 7 dose is calculated via a weighted average using the frequencies provided in Reference 19. The large isolation failure dose in Reference 19 is used as the Class 2 dose; the small isolation failure is used as the Class 6 dose. Reference 19 provides ISLOCA and SGTR doses; the Class 8 dose used in this analysis is weighted via the ISLOCA and SGTR frequencies in this calculation. Reference 19 provides the population dose (person-rem) for Class 1; 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

Accident Class	Description	Release (person-rem)
1	Containment Remains Intact	1.97E+03
2	Containment Isolation Failures	2.18E+05
3a	Independent or Random Isolation Failures SMALL	1.97E+04[1]
3b	Independent or Random Isolation Failures LARGE	1.97E+05[2]
4	Isolation Failure in which pre-existing leakage is not dependent on sequence progression. Type B test Failures	n/a
5	Isolation Failure in which pre-existing leakage is not dependent on sequence progression. Type C test Failures	n/a
6	Isolation Failure that can be verified by IST/IS or surveillance	8.92E+04
7	Containment Failure induced by severe accident	3.11E+05
8	Accidents in which containment is by-passed	4.76E+06

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

Release Category Definitions

Table 5-4 defines the accident classes used in the ILRT extension evaluation, which is consistent with the EPRI methodology [Reference 2]. These containment failure classifications are used in this analysis to determine the risk impact of extending the Containment Type A test interval, as described in Section 5.2 of this report.

Table 5-4 – EPRI Containment Failure Classification [Reference 2]

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 "large" failures in 217 tests (i.e., 2 / 217 = 0.0092). For Class 3b, the probability is based on the Jeffreys non-informative prior (i.e., 0.5 / 218 = 0.0023).

In a follow-up letter [Reference 20] to their ILRT guidance document [Reference 3], NEI issued additional information concerning the potential that the calculated delta LERF values for several plants may fall above the "very small change" guidelines of the NRC Regulatory Guide 1.174 [Reference 4]. This additional NEI information includes a discussion of conservatisms in the quantitative guidance for 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 MNS, as detailed in Section 5.2, involves subtracting Class 2 and Class 8 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 Report No. 1009325, Revision 2-A, Appendix H [Reference 24], EPRI TR-104285 [Reference 2] and previous risk assessment submittals on this subject [References 5, 8, 21, 22, and 23] have led to the following results. The results are displayed according to the eight accident classes defined in the EPRI report, as described in Table 5-5.

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

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

Table 5-5 – 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 (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 TR-104285 [Reference 2].) The question on containment integrity was modified to include the probability of a liner breach or bellows failure (due to excessive leakage) at the time of core damage. Two failure modes were considered for the Class 3 sequences. These are Class 3a (small breach) and Class 3b (large breach).

The frequencies for the severe accident classes defined in Table 5-5 were developed for MNS 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 presents the frequency and EPRI category for each sequence and the totals of each EPRI classification. Table 5-8 provides a summary of the accident sequence frequencies that can lead to radionuclide release to the public and have been derived consistent with the definitions of accident classes defined in EPRI TR-104285 [Reference 2], the NEI Interim Guidance [Reference 3], and guidance provided in EPRI Report No. 1009325, Revision 2-A [Reference 24]. Adjustments were made to the Class 3b and hence Class 1 frequencies to account for the impact of undetected corrosion of the steel liner per the methodology described in Section 5.2.6.

Note: calculations were performed with more digits than shown in this section. Therefore, minor differences may occur if the calculations in these sections are followed explicitly.

The total CDF (excluding high winds) is 1.34E-5 and LERF is 9.12E-7 [Reference 17]. The current PRA model (Revision 4a) does not contain a full Level 2 PRA, so some conservative assumptions are made when classifying risk.

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. The LERF model meets Capability Category I of the PRA standard, which is acceptable for the ILRT application (see Section A.2). However, this means the model may be conservative. Subtracting a conservative LERF value from CDF when calculating the Class 3B frequency would result in a non-conservative value. Therefore, only the large containment isolation (Class 2) and containment bypass (Class 8) contributions to LERF are removed from CDF, since these frequencies are not expected to be significantly influenced by potential conservatisms in the LERF model. The frequency of a Class 3a failure is calculated by the following equation:

$$Freq_{class3a} = P_{class3a} * (CDF - Class\ 2 - Class\ 8)$$
$$= \frac{2}{217} * (1.34\text{E-}5 - 3.98\text{E-}9 - 3.90\text{E-}7) = 1.20\text{E-}7$$

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:

$$Freq_{class3b} = P_{class3b} * (CDF - Class\ 2 - Class\ 8)$$
$$= \frac{.5}{218} * (1.34\text{E-}5 - 3.98\text{E-}9 - 3.90\text{E-}7) = 2.99\text{E-}8$$

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). Since the PRA model does not contain a Level 2 model, Class 1 is estimated as CDF – LERF. The frequency per year is initially determined from the EPRI Accident Class 1 frequency listed in Table 5-7 and then subtracting the EPRI Class 3a and 3b frequency (to preserve total CDF), calculated below:

$$Freq_{class1} = Freq_{class1} - (Freq_{class3a} - Freq_{class3b})$$

Class 2 Sequences. This group consists of core damage accident progression bins with large containment isolation failures. This is determined from summing TAG-ZL and %RPV, the contribution of large containment isolation failure flag and reactor pressure vessel failure, respectively, for LERF. Since these events are in cutsets that contribute 0.436% of LERF, which is 9.12E-7, the Class 2 contribution is 3.98E-9. The frequency per year for these sequences is obtained from the EPRI Accident Class 2 frequency listed in Table 5-7.

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 5-7.

Class 8 Sequences. This group consists of all core damage accident progression bins in which containment is bypassed via ISLOCA or SGTR. Since the ISLOCA initiator is in cutsets that contribute 0.047% of LERF, its Class 8 contribution is 6.34E-9. Since the SGTR initiators are in cutsets that contribute 2.86% of LERF, its Class 8 contribution is 3.84E-07. For this analysis, the frequency is determined from the EPRI Accident Class 8 frequency listed in Table 5-7.

LERF quantification is distributed into EPRI categories based on release categories. Table 5-6 shows this distribution.

Table 5-6 – Release Category Frequencies

Containment End State	EPRI Category	Frequency (/yr)
Intact Containment	1	1.25E-05
Large Isolation Failure	2	3.98E-09
Failures Induced by Phenomena	7	5.18E-07
ISLOCA	8	6.34E-09
SGTR	8	3.84E-07

Table 5-7 – Accident Class Frequencies

EPRI Category	Frequency (/yr)
Class 1	1.25E-05
Class 2	3.98E-09
Class 6	0
Class 7	5.18E-07
Class 8	3.90E-07
Total (CDF)	1.34E-05

Table 5-8 – Baseline Risk Profile

Class	Description	Frequency (/yr)
1	No containment failure	1.24E-05[2]
2	Large containment isolation failures	3.98E-09
3a	Small isolation failures (liner breach)	1.20E-07
3b	Large isolation failures (liner breach)	2.99E-08
4	Small isolation failures - failure to seal (type B)	ε[1]
5	Small isolation failures - failure to seal (type C)	ε[1]
6	Containment isolation failures (dependent failure, personnel errors)	ε[1]
7	Severe accident phenomena induced failure (early and late)	5.18E-07
8	Containment bypass	3.90E-07
	Total	1.34E-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 EPRI accident class. Table 5-9 provides a correlation of MNS 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:

$$EPRI\ Class\ 3a\ Population\ Dose = 10 * 1.97E{+}3 = 1.97E{+}4$$

$$EPRI\ Class\ 3b\ Population\ Dose = 100 * 1.97E{+}3 = 1.97E{+}5$$

Table 5-9 – Mapping of Population Dose to EPRI Accident Class

EPRI Category	Frequency (/yr)	Dose (person-rem)
Class 1	1.25E-05	1.97E+03
Class 2	3.98E-09	2.18E+05
Class 7	5.18E-07	3.11E+05
Class 8	3.90E-07	4.76E+06

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

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

Risk Impact Due to 10-Year Test Interval

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

$$Freq_{Class3a10yr} = \frac{10}{3} * \frac{2}{217} * (CDF - Class\ 2 - Class\ 8) = \frac{10}{3} * \frac{2}{217} * 1.30\text{E-}5 = 4.01\text{E-}7$$

$$Freq_{Class3b10yr} = \frac{10}{3} * \frac{.5}{218} * (CDF - Class\ 2 - Class\ 8) = \frac{10}{3} * \frac{.5}{218} * 1.30\text{E-}5 = 9.97\text{E-}8$$

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

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

Class	Description	Frequency (/yr)	Contribution (%)	Population Dose (person-rem)	Population Dose Rate (person-rem/yr)
1	No containment failure[2]	1.20E-05	89.49%	1.97E+03	2.37E-02
2	Large containment isolation failures	3.98E-09	0.03%	2.18E+05	8.67E-04
3a	Small isolation failures (liner breach)	4.01E-07	2.98%	1.97E+04	7.90E-03
3b	Large isolation failures (liner breach)	9.97E-08	0.74%	1.97E+05	1.96E-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]	8.92E+04	ε[1]
7	Severe accident phenomena induced failure (early and late)	5.18E-07	3.85%	3.11E+05	1.61E-01
8	Containment bypass	3.90E-07	2.91%	4.76E+06	1.86E+00
	Total	1.34E-05			2.07E+00

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

$$Freq_{Class3a15yr} = \frac{15}{3} * \frac{2}{217} * (CDF - Class\ 2 - Class\ 8) = 5 * \frac{2}{217} * 1.30\text{E-}5 = 6.01\text{E-}7$$

$$Freq_{Class3b15yr} = \frac{15}{3} * \frac{.5}{218} * (CDF - Class\ 2 - Class\ 8) = 5 * \frac{.5}{218} * 1.30\text{E-}5 = 1.50\text{E-}7$$

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

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

Class	Description	Frequency (/yr)	Contribution (%)	Population Dose (person-rem)	Population Dose Rate (person-rem/yr)
1	No containment failure[2]	1.18E-05	87.85%	1.97E+03	2.32E-02
2	Large containment isolation failures	3.98E-09	0.03%	2.18E+05	8.67E-04
3a	Small isolation failures (liner breach)	6.01E-07	4.47%	1.97E+04	1.18E-02
3b	Large isolation failures (liner breach)	1.50E-07	1.11%	1.97E+05	2.95E-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]	8.92E+04	ε[1]
7	Severe accident phenomena induced failure (early and late)	5.18E-07	3.85%	3.11E+05	1.61E-01
8	Containment bypass	3.90E-07	2.91%	4.76E+06	1.86E+00
	Total	1.34E-05			2.08E+00

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

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

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

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

at MNS, the ILRT extension does not impact CDF. Therefore, the relevant risk-impact metric is LERF.

For MNS, 100% of the frequency of Class 3b sequences can be used as a very conservative first-order estimate to approximate the potential increase in LERF from the ILRT interval extension (consistent with the EPRI guidance methodology). Based on a 10-year test interval from Table 5-10, the Class 3b frequency is 9.97E-8/year; based on a 15-year test interval from Table 5-11, the Class 3b frequency is 1.50E-7/year. 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 1.20E-7/year. Similarly, the increase due to increasing the interval from 10 to 15 years is 4.99E-8/year. As can be seen, even with the conservatisms included in the evaluation (per the EPRI methodology), the estimated change in LERF is below the threshold criteria for a small change when comparing the 15-year results to the current 10-year requirement, and slightly greater than the criteria when compared to the original 3-year requirement. Table 5-12 summarizes these results.

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

ILRT Inspection Interval	3 Years (baseline)	10 Years	15 Years
Class 3b (Type A LERF)	2.99E-08	9.97E-08	1.50E-07
ΔLERF (3 year baseline)		6.98E-08	1.20E-07
ΔLERF (10 year baseline)			4.99E-08

The increase in the overall probability of LERF due to Class 3b sequences is slightly greater than 10^{-7}. As stated in RG 1.174 [Reference 4], "When the calculated increase in LERF is in the range of 10^{-7} per reactor year to 10^{-6} per reactor year, applications will be considered only if it can be reasonably shown that the total LERF is less than 10^{-5} per reactor year." Baseline Internal Events LERF is 9.12E-7. Therefore, there is significant margin for both the ΔLERF and baseline LERF to the upper limits of Region II in RG 1.174 [Reference 4].

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

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

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

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

Table 5-13 shows the steps and results of this calculation. The difference in CCFP between the 3-year test interval and 15-year test interval is 0.891%.

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

ILRT Inspection Interval	3 Years (baseline)	10 Years	15 Years
f(ncf) (/yr)	1.25E-05	1.24E-05	1.24E-05
f(ncf)/CDF	0.930	0.925	0.921
CCFP	0.0701	0.0753	0.0790
ΔCCFP (3 year baseline)		0.519%	0.891%
ΔCCFP (10 year baseline)			0.371%

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 0.891%. Therefore, this increase is judged to be very 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 5-14, 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 McGuire has a different containment type, this event could potentially occur at McGuire (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. McGuire has a free standing steel containment and would not be subject to the same type of event. Therefore, this event does not apply to McGuire. 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 is one corrosion event from the 5.5 years that applies to McGuire. The Brunswick corrosion event could potentially occur in any

containment [Reference 42].

- The McGuire containment shell is much thicker than the containment liners of steel and concrete containments. Since the steel containment is much thicker than the typical steel liner of a concrete containment, it will take longer for a through wall hole to develop due to corrosion. Additionally, construction material in the concrete containment next to the liner may have contributed to the Brunswick event. Therefore, to account for these differences, a factor of 0.1 will be applied to the containment cylinder and dome value. No additional credit will be given to the containment basemat since the portion of containment embedded in the basemat is more similar to other containment types compared to the containment cylinder and dome [Reference 42].
- 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 5-14, 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 MNS, the ILRT maximum pressure is 15.2 psig [References 27]. 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 5.3.2 (See Table 5-14, 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 5-14, Step 4).
- Consistent with the Calvert Cliffs analysis, a 5% visual inspection detection failure likelihood given the flaw is visible and a total detection failure likelihood of 10% is used. To date, all liner corrosion events have been detected through visual inspection (See Table 5-14, 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 5-14 – Steel Liner Corrosion Base Case

Step	Description	Containment Cylinder and Dome (82%)		Containment Basemat (18%)	
1	Historical liner flaw likelihood Failure data: containment location specific Success data: based on 70 steel-lined containments and 5.5 years since the 10CFR 50.55a requirements of periodic visual inspections of containment surfaces	Events: 1 (Brunswick 2) 1 / (70 x 5.5) * 0.1 = 2.60E-04		Events: 0 Assume a half failure 0.5 / (70 x 5.5) = 1.30E-03	
2	Aged adjusted liner flaw likelihood During the 15-year interval, assume failure rate doubles every five years (14.9% increase per year). The average for the 5th to 10th year set to the historical failure rate.	Year	Failure rate	Year	Failure rate
		1	1.03E-04	1	5.13E-04
		average 5-10	2.60E-04	average 5-10	1.30E-03
		15	7.14E-04	15	3.57E-03
		15 year average = 3.22E-04		15 year average = 1.61E-03	
3	Increase in flaw likelihood between 3 and 15 years Uses aged adjusted liner flaw likelihood (Step 2), assuming failure rate doubles every five years.	0.036% (1 to 3 years) 0.21% (1 to 10 years) 0.48% (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)	0.000036% (3 years) 0.036% x 1% x 10% 0.00021% (10 years) 0.21% x 1% x 10% 0.000480% (15 years) 0.48% 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 MNS.

Table 5-15 – Total Likelihood on Non-Detected Containment Leakage Due to Corrosion for MNS

Description
At 3 years: 0.000036% + 0.00018% = 0.00021%
At 10 years: 0.00021% + 0.00104% = 0.00124%
At 15 years: 0.00480% + 0.00242% = 0.00290%

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. 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 recenly, 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 28].

5.3 Sensitivities

5.3.1 Potential 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.

McGuire has transitioned to NFPA 805 licensing basis for fire protection and submitted a License Amendment Request (LAR) [Reference 39]. This transition included performing a Fire PRA and installing modifications to reduce the fire-induced CDF and LERF to those reported in the NFPA 805 LAR; all committed plant modifications have been completed.

The Fire PRA model fire-mr3a_r2v24 was used to obtain the fire CDF and LERF values [Reference 18]. As described in Section 5.1.3, additional consideration is made to not apply failure probabilities on those cases that are already LERF scenarios. Therefore, LERF contributions from CDF are removed. The following shows the calculation for Class 3b:

$$Freq_{U1class3b} = P_{class3b} * (CDF - LERF) = \frac{0.5}{218} * (3.45E\text{-}5 - 4.30E\text{-}6) = 6.92E\text{-}8$$

$$Freq_{U2class3b} = P_{class3b} * (CDF - LERF) = \frac{0.5}{218} * (4.17E\text{-}5 - 5.04E\text{-}6) = 8.40E\text{-}8$$

$$Freq_{U1class3b10yr} = \frac{10}{3} * P_{class3b} * (CDF - LERF) = \frac{10}{3} * \frac{0.5}{218} * (3.45E\text{-}5 - 4.30E\text{-}6) = 2.31\text{E-}7$$

$$Freq_{U2class3b10yr} = \frac{10}{3} * P_{class3b} * (CDF - LERF) = \frac{10}{3} * \frac{0.5}{218} * (4.17E\text{-}5 - 5.04E\text{-}6) = 2.80\text{E-}7$$

$$Freq_{U1class3b15yr} = \frac{15}{3} * P_{class3b} * (CDF - LERF) = 5 * \frac{0.5}{218} * (3.45E\text{-}5 - 4.30E\text{-}6) = 3.46\text{E-}7$$

$$Freq_{U1class3b15yr} = \frac{15}{3} * P_{class3b} * (CDF - LERF) = 5 * \frac{0.5}{218} * (4.17E\text{-}5 - 5.04E\text{-}6) = 4.20\text{E-}7$$

The 2014 Seismic Reevaluations for operating reactor sites [Reference 41] states the conclusions reached in 2010 by GI-199 [Reference 32] remain valid for estimating Seismic CDF at plants in the Central and Eastern United States, which includes MNS. EPRI guidance [Reference 35] on recent seismic evaluations states, "EPRI does not recommend using any very conservative approaches to estimate the SCDF such as use of the maximum SCDFs calculated at any one frequency. This type of bounding approach is overly conservative and judged to not provide realistic risk estimates consistent with SCDFs calculated in actual SPRAs." Therefore, the average of the McGuire Seismic CDF values reported in Table D-1 of GI-199 [Reference 32] is calculated as follows:

$$\text{CDF}_{\text{Seismic}} = (3.1\text{E-}5 + 3.0\text{E-}05 + 1.5\text{E-}05 + 1.1\text{E-}05)/4 = 2.18\text{E-}5$$

Applying the internal event LERF/CDF ratio to the seismic CDF yields an estimated seismic LERF of 1.48E-6, as shown by the equation below.

$$\text{LERF}_{\text{Seismic}} \approx \text{CDF}_{\text{Seismic}} * \text{LERF}_{\text{IE}} / \text{CDF}_{\text{IE}} = 2.18\text{E-}5 * 9.12\text{E-}7 / 1.34\text{E-}5 = 1.48\text{E-}6$$

To reduce conservatism in the model, the methodology of subtracting existing LERF from CDF is applied to the Seismic PRA model. The following shows the calculation for Class 3b:

$$Freq_{class3b} = P_{class3b} * (CDF - LERF) = \frac{0.5}{218} * (2.18E\text{-}5 - 1.48E\text{-}6) = 4.65\text{E-}8$$

$$Freq_{class3b10yr} = \frac{10}{3} * P_{class3b} * (CDF - LERF) = \frac{10}{3} * \frac{0.5}{218} * (2.18E\text{-}5 - 1.48E\text{-}6) = 1.55\text{E-}7$$

$$Freq_{class3b15yr} = \frac{15}{3} * P_{class3b} * (CDF - LERF) = 5 * \frac{0.5}{218} * (2.18E\text{-}5 - 1.48E\text{-}6) = 2.32\text{E-}7$$

The HWPRA contains some event probabilities and wind fragilities near or equal to 1.0. This has a conservative effect on CDF and LERF because CAFTA uses the min-cut-upper-bound approach to estimate the mean probability from cutsets [Reference 36]. The CDF and LERF of the Rev. 4a model were solved for an exact solution using ACUBE to address the limitations in the min-cut-upper-bound approach: CDF_{HW} = 8.65E-6, LERF_{HW} = 8.67E-7. To reduce conservatism in the analysis, the methodology of subtracting existing LERF from CDF is applied to the HWPRA model. The following shows the calculation for Class 3b:

$$Freq_{class3b} = P_{class3b} * (CDF - LERF) = \frac{0.5}{218} * (8.65E\text{-}6 - 8.67E\text{-}7) = 1.79\text{E-}8$$

$$Freq_{class3b10yr} = \frac{10}{3} * P_{class3b} * (CDF - LERF) = \frac{10}{3} * \frac{0.5}{218} * (8.65E\text{-}6 - 8.67E\text{-}7) = 5.95\text{E-}8$$

$$Freq_{class3b15yr} = \frac{15}{3} * P_{class3b} * (CDF - LERF) = 5 * \frac{0.5}{218} * (8.65E\text{-}6 - 8.67E\text{-}7) = 8.93\text{E-}8$$

The IPEEE reported an external flood CDF of 5.0E-9. Many other external events were reviewed and screened [Reference 34]. As with the other models, the internal event LERF/CDF ratio to the seismic CDF yields an estimated seismic LERF of 3.39E-10, and the methodology of subtracting existing LERF from CDF is applied to the HWPRA model. The following shows the calculation for Class 3b:

$$Freq_{class3b} = P_{class3b} * (CDF - LERF) = \frac{0.5}{218} * (5.0E\text{-}9 - 3.39E\text{-}10) = 1.07\text{E-}11$$

$$Freq_{class3b10yr} = \frac{10}{3} * P_{class3b} * (CDF - LERF) = \frac{10}{3} * \frac{0.5}{218} * (5.0E\text{-}9 - 3.39E\text{-}10) = 3.56\text{E-}11$$

$$Freq_{class3b15yr} = \frac{15}{3} * P_{class3b} * (CDF - LERF) = 5 * \frac{0.5}{218} * (5.0E\text{-}9 - 3.39E\text{-}10) = 5.34\text{E-}11$$

The fire, seismic, high winds, and external flood 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 5-16.

Table 5-16 – Unit 1 MNS 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	1.34E-07	4.45E-07	6.68E-07	5.34E-07
Internal Events	2.99E-08	9.97E-08	1.50E-07	1.20E-07
Combined	1.63E-07	5.45E-07	8.17E-07	6.54E-07

Table 5-17 – Unit 2 MNS 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	1.48E-07	4.95E-07	7.42E-07	5.94E-07
Internal Events	2.99E-08	9.97E-08	1.50E-07	1.20E-07
Combined	1.78E-07	5.94E-07	8.92E-07	7.13E-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 2.72E-7 for Unit 1 and 2.97E-7 for Unit 2; the total change in LERF due to increasing the ILRT interval from 3 to 15 years is 6.54E-7 for Unit 1 and 7.13E-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 values are calculated below:

$$\text{LERF} = \text{LERF}_{\text{internal}} + \text{LERF}_{\text{fire}} + \text{LERF}_{\text{seismic}} + \text{LERF}_{\text{HW}} + \text{LERF}_{\text{ExFlood}} + \text{LERF}_{\text{class3Bincrease}}$$

$$\text{LERF}_{\text{U1-10yr}} = 9.12\text{E-}7/\text{yr} + 4.30\text{E-}6/\text{yr} + 1.48\text{E-}6/\text{yr} + 8.67\text{E-}7/\text{yr} + 3.39\text{E-}10/\text{yr} + 2.72\text{E-}7/\text{yr} = 7.83\text{E-}6/\text{yr}$$

$$\text{LERF}_{\text{U2-10yr}} = 9.12\text{E-}7/\text{yr} + 5.04\text{E-}6/\text{yr} + 1.48\text{E-}6/\text{yr} + 8.67\text{E-}7/\text{yr} + 3.39\text{E-}10/\text{yr} + 2.97\text{E-}7/\text{yr} = 8.59\text{E-}6/\text{yr}$$

$$\text{LERF}_{\text{U1-15yr}} = 9.12\text{E-}7/\text{yr} + 4.30\text{E-}6/\text{yr} + 1.48\text{E-}6/\text{yr} + 8.67\text{E-}7/\text{yr} + 3.39\text{E-}10/\text{yr} + 6.54\text{E-}7/\text{yr} = 8.21\text{E-}6/\text{yr}$$

$$\text{LERF}_{\text{U2-15yr}} = 9.12\text{E-}7/\text{yr} + 5.04\text{E-}6/\text{yr} + 1.48\text{E-}6/\text{yr} + 8.67\text{E-}7/\text{yr} + 3.39\text{E-}10/\text{yr} + 7.13\text{E-}7/\text{yr} = 9.01\text{E-}6/\text{yr}$$

As specified in Regulatory Guide 1.174 [Reference 4], since the total LERF is less than 1.0E-5, it is acceptable for the ΔLERF to be between 1.0E-7 and 1.0E-6.

5.3.2 Potential Impact from Steel Liner Corrosion Likelihood

A quantitative assessment of the contribution of steel liner corrosion likelihood impact was performed for the risk impact assessment for extended ILRT intervals. As a sensitivity run, the internal event CDF was used to calculate the Class 3b frequency. The impact on the Class 3b frequency due to increases in the ILRT surveillance interval was calculated for steel liner corrosion likelihood using the relationships described in Section 5.2.6. The EPRI Category 3b frequencies for the 3 per 10-year, 10-year, and 15-year ILRT intervals were quantified using the internal events CDF. The change in the LERF, change in CCFP, and change in Annual Dose Rate due to extending the ILRT interval from 3 in 10 years to 1 in 10 years, or to 1 in 15 years are provided in Table 5-18 – Table 5-20. The steel liner corrosion likelihood was increased by a factor of 1000, 10000, and 100000. Except for extreme factors of 10000 and 100000, the corrosion likelihood is relatively insensitive to the results.

Table 5-18 – Steel Liner Corrosion Sensitivity Case: 3B Contribution

	3b Frequency (3-per-10 year ILRT)	3b Frequency (1-per-10 year ILRT)	3b Frequency (1-per-15 year ILRT)	LERF Increase (3-per-10 to 1-per-10)	LERF Increase (3-per-10 to 1-per-15)	LERF Increase (1-per-10 to 1-per-15)
Internal Event 3B Contribution	5.36E-08	1.79E-07	2.68E-07	1.25E-07	2.14E-07	8.93E-08
Corrosion Likelihood X 1000	5.37E-08	1.81E-07	2.76E-07	1.27E-07	2.22E-07	9.49E-08
Corrosion Likelihood X 10000	5.47E-08	2.01E-07	3.46E-07	1.46E-07	2.91E-07	1.45E-07
Corrosion Likelihood X 100000	6.51E-08	4.01E-07	1.04E-06	3.36E-07	9.80E-07	6.44E-07

Table 5-19 – Unit 1 Steel Liner Corrosion Sensitivity: CCFP

	CCFP (3-per-10 year ILRT)	CCFP (1-per-10 year ILRT)	CCFP (1-per-15 year ILRT)	CCFP Increase (3-per-10 to 1-per-10)	CCFP Increase (3-per-10 to 1-per-15)	CCFP Increase (1-per-10 to 1-per-15)
Baseline CCFP	7.01E-02	7.53E-02	7.90E-02	5.19E-03	8.91E-03	3.71E-03
Corrosion Likelihood X 1000	7.01E-02	7.53E-02	7.90E-02	5.21E-03	8.92E-03	3.72E-03
Corrosion Likelihood X 10000	7.02E-02	7.55E-02	7.92E-02	5.31E-03	9.10E-03	3.79E-03
Corrosion Likelihood X 100000	7.06E-02	7.69E-02	8.14E-02	6.30E-03	1.08E-02	4.50E-03

Table 5-20 – Unit 1 Steel Liner Corrosion Sensitivity: Dose Rate

	Dose Rate (3-per-10 year ILRT)	Dose Rate (1-per-10 year ILRT)	Dose Rate (1-per-15 year ILRT)	Dose Rate Increase (3-per-10 to 1-per-10)	Dose Rate Increase (3-per-10 to 1-per-15)	Dose Rate Increase (1-per-10 to 1-per-15)
Dose Rate	1.43E-02	4.76E-02	7.14E-02	3.33E-02	5.71E-02	2.38E-02
Corrosion Likelihood X 1000	1.43E-02	4.77E-02	7.15E-02	3.34E-02	5.72E-02	2.38E-02
Corrosion Likelihood X 10000	1.46E-02	4.86E-02	7.29E-02	3.40E-02	5.83E-02	2.43E-02
Corrosion Likelihood X 100000	1.73E-02	5.77E-02	8.66E-02	4.04E-02	6.93E-02	2.89E-02

5.3.3 Expert Elicitation Sensitivity

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

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

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

The baseline assessment uses the Jeffreys non-informative prior and the expert elicitation sensitivity study uses the results of the expert elicitation. In addition, given the relationship between leakage magnitude and probability, larger leakage that is more representative of large early release frequency, can be reflected. For the purposes of this sensitivity, the same leakage magnitudes that are used in the basic methodology (i.e., 10 L_a for small and 100 L_a for large) are used here. Table 5-21 presents the magnitudes and probabilities associated with the Jeffreys non-informative prior and the expert elicitation used in the base methodology and this sensitivity case.

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

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

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

Table 5-22 – MNS Summary of ILRT Extension Using Expert Elicitation Values

Accident Class	ILRT Interval							
	3 per 10 Years				1 per 10 Years		1 per 15 Years	
	Base Frequency	Adjusted Base Frequency	Dose (person-rem)	Dose Rate (person-rem/yr)	Frequency	Dose Rate (person-rem/yr)	Frequency	Dose Rate (person-rem/yr)
1	1.25E-05	1.24E-05	1.97E+03	2.45E-02	1.22E-05	2.40E-02	1.20E-05	2.37E-02
2	3.98E-09	3.98E-09	2.18E+05	8.67E-04	3.98E-09	8.67E-04	3.98E-09	8.67E-04
3a	N/A	9.07E-08	1.97E+04	1.79E-03	3.02E-07	5.95E-03	4.53E-07	8.93E-03
3b	N/A	5.77E-09	1.97E+05	1.14E-03	1.92E-08	3.79E-03	2.89E-08	5.69E-03
7	5.18E-07	5.18E-07	3.11E+05	1.61E-01	5.18E-07	1.61E-01	5.18E-07	1.61E-01
8	3.90E-07	3.90E-07	4.76E+06	1.86E+00	3.90E-07	1.86E+00	3.90E-07	1.86E+00
Totals	1.34E-05	1.34E-05	5.50E+06	2.05E+00	1.34E-05	2.05E+00	1.34E-05	2.06E+00
ΔLERF (3 per 10 yrs base)	N/A				1.35E-08		2.31E-08	
ΔLERF (1 per 10 yrs base)	N/A				N/A		9.62E-09	
CCFP	6.83%				6.93%		7.00%	

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

5.3.4 Recovery Actions Sensitivity

The basis for the internal events recovery rules associated with removal of mutually exclusive combinations or combinations of events considered invalid was considered not well documented (see F&O 1-21 in Section A.1 for more details). Due to uncertainty in the basis for the mutually exclusive event recovery rules, the rules were reviewed and those that were judged to not have an adequately documented basis were identified. The rules considered to have inadequate documentation of basis were removed from the recovery rule file, and the internal events model was re-quantified with the modified recovery rule to asses any potential risk increase that would result from inappropriate mutually exclusive combination removal.

The results of this sensitivity study show that CDF increases by 5.21E-07, Class 2 LERF increases by 4.91E-11, Class 8 LERF increases by 1.16E-8, and total LERF increases by 2.08E-8. The ΔLERF due to increasing the ILRT test interval from 3 in 10 to 1 in 15 years is 1.24E-7/year for this sensitivity. Results are shown in Table 5-23. These ΔLERF results are similar to the results quantified using the model that includes the mutually exclusive event recovery rules, showing that the open F&O has no impact on the conclusion of the ILRT extension risk analysis.

Table 5-23 – Impact on LERF due to Extended Type A Testing Intervals for Recovery Rules Sensitivity

ILRT Inspection Interval	3 Years (baseline)	10 Years	15 Years
Class 3b (Type A LERF)	3.11E-08	1.04E-07	1.55E-07
ΔLERF (3 year baseline)		7.25E-08	1.24E-07
ΔLERF (10 year baseline)			5.18E-08

Total internal events LERF is 1.06E-6. As specified in Regulatory Guide 1.174 [Reference 4], since the total LERF is less than 1.0E-5, it is acceptable for the ΔLERF to be between 1.0E-7 and 1.0E-6.

5.3.5 Alternate Equipment Alignment Sensitivity

For systems that are normally running, the MNS model generally does not assume alternate train alignments and instead assumes the same train is always in service. Therefore some failure modes were not evaluated for those trains such as start failures, unavailability, and the impact of pre-initiators (see F&O 5-5 in Section A.1 for more details). Not modeling all failure modes potentially underestimates risk, which is non-conservative for this ILRT application. In lieu of a detailed systems analysis for alternate alignments that may introduce additional failure modes, a very conservative sensitivity is performed where the internal events (excluding internal floods) CDF and LERF were doubled to easily bound the potential added risk of including all failure modes. 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 1.59E-7/year for this sensitivity. Results are shown in Table 5-24. These ΔLERF results are similar to the baseline results in Section 5.2.4, showing that the open F&O has no impact on the conclusion of the ILRT extension risk analysis.

Table 5-24 – Impact on LERF due to Extended Type A Testing Intervals for Equipment Alignment Sensitivity

ILRT Inspection Interval	3 Years (baseline)	10 Years	15 Years
Class 3b (Type A LERF)	3.96E-08	1.32E-07	1.98E-07
ΔLERF (3 year baseline)		9.25E-08	1.59E-07
ΔLERF (10 year baseline)			6.61E-08

Total internal events LERF is 1.58E-6. As specified in Regulatory Guide 1.174 [Reference 4], since the total LERF is less than 1.0E-5, it is acceptable for the ΔLERF to be between 1.0E-7 and 1.0E-6.

6.0 RESULTS

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

Table 6-1 – 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	1.97E+03	1.24E-05	2.44E-02	1.20E-05	2.37E-02	1.18E-05	2.32E-02
2	2.18E+05	3.98E-09	8.67E-04	3.98E-09	8.67E-04	3.98E-09	8.67E-04
3a	1.97E+04	1.20E-07	2.37E-03	4.01E-07	7.90E-03	6.01E-07	1.18E-02
3b	1.97E+05	2.99E-08	5.89E-03	9.97E-08	1.96E-02	1.50E-07	2.95E-02
7	3.11E+05	5.18E-07	1.61E-01	5.18E-07	1.61E-01	5.18E-07	1.61E-01
8	4.76E+06	3.90E-07	1.86E+00	3.90E-07	1.86E+00	3.90E-07	1.86E+00
Total		**1.34E-05**	**2.05E+00**	**1.34E-05**	**2.07E+00**	**1.34E-05**	**2.08E+00**

ILRT Dose Rate from 3a and 3b				
ΔTotal Dose Rate	From 3 Years	N/A	1.86E-02	3.19E-02
	From 10 Years	N/A	N/A	1.33E-02
%ΔDose Rate	From 3 Years	N/A	0.91%	1.55%
	From 10 Years	N/A	N/A	0.64%
3b Frequency (LERF)				
ΔLERF	From 3 Years	N/A	6.98E-08	1.20E-07
	From 10 Years	N/A	N/A	4.99E-08
CCFP %				
ΔCCFP%	From 3 Years	N/A	0.519%	0.891%
	From 10 Years	N/A	N/A	0.371%

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 1.20E-7/year 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. Baseline LERF is 9.12E-07. 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. 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 4.99E-08, 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 6.54E-7/year for Unit 1 and 7.13E-7/year for Unit 2 using the EPRI guidance, and baseline LERF is 8.21E-6/year for Unit 1 and 9.01E-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 2.72E-07 for Unit 1 and 2.97E-07 for Unit 2. When external event risk is included, total LERF is 7.83E-06 for Unit 1 and 8.59E-06 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 5.3.1, the EPRI methodology used to estimate the increase in LERF is conservative. Therefore, even though the increase in 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 0.032 person-rem/year. EPRI Report No. 1009325, Revision 2-A [Reference 24] states that a very small population dose is defined as an increase of ≤ 1.0 person-rem per year, or ≤ 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 0.891%. EPRI Report No. 1009325, Revision 2-A [Reference 24] states that increases in CCFP of ≤ 1.5% is very small. Therefore, this increase is judged to be very small.

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

A. ATTACHMENT 1

PRA Quality Statement for Permanent 15-Year ILRT Extension

The internal flood PRA model received a focused peer review against the ASME/ANS RA-Sa-2009 PRA Standard in September, 2011, while the LERF PRA model received a focused peer review against the ASME/ANS RA-Sa-2009 PRA Standard in December, 2012. The McGuire internal events model (excluding LERF) received a peer review against the requirements of the ASME/ANS RA-Sa-2009 PRA Standard, using the process defined in Nuclear Energy Institute (NEI) 05-04 in June, 2015. The scope of this June, 2015 Peer Review included all internal event requirements, as well as a closure review of the resolutions to findings against the LERF and Internal Flooding models. The technical adequacy of the analysis performed to resolve all finding-level Facts and Observations (F&Os) of the internal flooding and LERF PRAs were reviewed against Capability Category II (CC II) supporting requirements. As a result, relatively few internal flooding and LERF finding-level F&Os remain open. Following the June, 2015 peer review, additional analysis was performed to address the Facts and Observations against the internal events (excluding LERF). The technical adequacy of this additional analysis was reviewed against the individual supporting requirement CC II by another independent third-party review team [Reference 37].

The McGuire Fire PRA model received a peer review against the requirements of the ASME/ANS RA-Sa-2009 PRA Standard, using NEI 07-12. The high winds PRA model received a peer review against the requirements of the ASME/ANS RA-Sb-2013 PRA Standard, using NEI 05-04, after a comparison between the ASME/ANS RA-Sa-2009 and ASME/ANS RA-Sb-2013 standards showed no substantive differences for high winds.

The PRA model is maintained and updated such that its representation of the as-built, as-operated plant is sufficient to support the applications for which it is used. Duke Energy maintains procedures that evaluate and prioritize changes in PRA inputs as well as address discovery of new information that could affect the PRA.

The PRA model is reviewed whenever plant accident response characteristics are changed. Any identifiable plant change is analyzed for its risk significance. This includes plant physical modifications, changes to Emergency or Abnormal Procedures, as well as Technical Specifications and Selected Licensee Commitment changes. Additionally, all open PRA Tracker items are reviewed prior to the start of an application for their impact on that application.

A.1. Internal Events CDF Model

Three (3) internal events PRA finding-level F&Os are considered to be open. Each of the open finding-level F&Os is discussed, as follows.

F&Os Regarding PRA Supporting Requirements		
F&O ID:	**Associated SRs:**	**Peer Review CC Assessment:**
1-21	QU-A5, QU-B7, QU-B8	Met at CC I-III
SR QU-A5 Capability Category I/II/III Requirements:		
INCLUDE recovery actions in the quantification process in applicable sequences and cutsets (see Requirements HR-H1, HR-H2 and HR-H3).		
F&O Issue and Proposed Resolution:		
The rationale for the recovery rules associated with removal of mutually exclusive combinations or combinations of events considered invalid was not well documented. The mutually exclusive maintenance rules reference Technical Specifications and administrative guidance in a general manner, but the specific justification for each combination is not clear. For combinations considered to be invalid for other reasons, notes are included but they are somewhat general in nature and do not provide a complete justification. For example: - The inverter function is only required early in the event. - An appropriately short mission time would cause these to be truncated. - An operator recovery could also be added. While the inverter is needed early for ESFAS initiation, it is also needed to support later functions like auto-alignment of the sump suction valves to ECCS. Therefore, it may be more appropriate to add the recovery HFE (e.g., SMAN001RHE) to these combinations rather than removing them to ensure that the later functions dependent on the inverters are recognized. Also, the rules for SMAN001RHE say: - These sequences involve small or medium LOCAs and a failure of the automatic swap of ECCS suction to the sump. It is not clear why this would not be valid for Large LOCA since transfer to recirculation would not occur until several minutes after a Large LOCA. If MAAP analysis shows that there is not sufficient time for this recovery with a Large LOCA, this should be stated with a reference to the appropriate MAAP case. (This F&O originated from SR QU-A5) **Basis for Significance:** The rationale for removal of cutsets from the quantification results should be thoroughly explained and appropriately justified so that the results can be verified without recourse to the author. **Possible Resolution:**		

Add a description of the mutually exclusive combinations to the system notebooks with clear justification (e.g., reference to the appropriate Technical Specification or administrative guidance).

For other rules used to remove cutsets considered to be invalid, either correct the logic to prevent generation of those combinations or provide a complete justification in the model integration notebook.

Disposition of the Peer Review Finding:

Reason F&O Remains Open following January, 2016 Independent Review of PRA:

The recovery rule file contains a significant amount of MTX style combinations without adequate documentation to justify. Some of the combinations appear to be used in place of improving the fault trees. If these are not well documented and reviewed when the models are updated there is potential that some cutsets may be inadvertently deleted. Suggest reviewing all combinations and revising the model if possible, or creating a MTX notebook to document the justifications.

Evaluation of F&O impact on proposed application:

The recovery rule file has been reviewed for potential impact on this application. Due to uncertainty in the basis for the mutually exclusive event recovery rules, the rules were reviewed and those that were judged to not have an adequately documented basis were identified. The rules considered to have inadequate documentation of basis were removed from the recovery rule file, and the internal events model was re-quantified with the modified recovery rule to asses any potential risk increase that would result from inappropriate mutually exclusive combination removal. This sensitivity is detailed in Section 5.3.4. As specified in Regulatory Guide 1.174 [Reference 4], since the total LERF is less than 1.0E-5, it is acceptable for the ΔLERF to be between 1.0E-7 and 1.0E-6.

F&Os Regarding PRA Supporting Requirements

F&O ID:	Associated SRs:	Peer Review CC Assessment:
2-7	SY-A5, SY-A19, DA-A2	Met at CC I-III

SR SY-A5 Capability Category I/II/III Requirements:

INCLUDE the effects of both normal and alternate system alignments, to the extent needed for CDF determination.

F&O Issue and Proposed Resolution:

Only one system alignment is included in the system models in most cases. The PACLOSS logic does include alternate alignment configurations for offsite power, but only one is used in the quantification.

(This F&O originated from SR SY-A5)

Basis for Significance:

A review of the systems shows that only normal operating alignments are typically included. The SR requires that "the effects of both normal and alternate system alignments, to the extent needed for CDF and LERF determination" be included. Since there is no documentation of the investigation of alternate alignments as required by the SR, it cannot be determined if there are unrecognized asymmetries that would be important to provide a full understanding of the CDF and LERF results.

Possible Resolution:

Include both normal and alternate system alignments in the system models. Quantify the model with alternate alignments to confirm there are no unrecognized asymmetries.

Disposition of the Peer Review Finding:

Reason F&O Remains Open following January, 2016 Independent Review of PRA:

Alternate alignments that are not normal have been addressed. However, in some systems there are alternative normal alignments that are not modeled. The model assumes one alignment as normal versus using split fraction for multiple normal alignments. This may be adequate for determining the base CDF, however some failure modes are not modeled which can create misleading insights in application space. (see also F&Os 4-4 and 5-5)

Split fractions can be applied for normal alignments or provide justification that the alignments modeled are adequate for the application the model will support.

The review team recognizes that there may be some interpretation differences regarding SR SY-A5 and notes that Duke has indicated it will be submitting an inquiry to the JCNRM for clarification.

Evaluation of F&O impact on proposed application:

Not modeling all system alignments potentially underestimates risk, which is non-conservative for this ILRT application. As a very conservative sensitivity, the internal events CDF (excluding internal floods) was doubled to easily bound the potential added risk of including all failure modes. Section 5.3.5 details this sensitivity. As specified in Regulatory Guide 1.174 [Reference 4], since the total LERF is less than 1.0E-5, it is acceptable for the ΔLERF to be between 1.0E-7 and 1.0E-6.

F&Os Regarding PRA Supporting Requirements

F&O ID:	Associated SRs:	Peer Review CC Assessment:
5-5	HR-A1	Met at CC I-III

Capability Category I/II/III Requirements:

For equipment modeled in the PRA, IDENTIFY through a review of procedures and practices, those test, inspection, and maintenance activities that require realignment of equipment outside its normal operational or standby status.

F&O Issue and Proposed Resolution:

All normally-running systems were excluded from consideration for pre-initiator HFE solely because they were operating. No evaluation that the function of normally running systems required to support the PRA was not affected by routine maintenance activities was performed.

(This F&O originated from SR HR-A1)

Basis for Significance:

This SR requires that all systems be evaluated.

Possible Resolution:

Identify the PRA mitigation function that is required of each normally-running system and how each potential failure mode that could be introduced by test, inspection, and maintenance activities is precluded by running the system in normal alignment.

Disposition of the Peer Review Finding:

Reason F&O Remains Open following January, 2016 Independent Review of PRA:

The MNS model generally assumes the same train is always running. Therefore some failure modes were not evaluated for those trains such as start failures, unavailability and the impact of pre-initiators. This could lead to inaccuracies when characterizing risk insights. (see also F&Os 2-7 and 4-4)

Evaluation of F&O impact on proposed application:

This concern applies to equipment that is modeled as running, but actually may be in a standby condition due to testing, maintenance, or operating the plant in an alternate alignment. Not modeling all failure modes potentially underestimates risk, which is non-conservative for this ILRT application. As a very conservative sensitivity, the internal events CDF (excluding internal floods) was doubled to easily bound the potential added risk of including all failure modes. Section 5.3.5 details this sensitivity. As specified in Regulatory Guide 1.174 [Reference 4], since the total LERF is less than 1.0E-5, it is acceptable for the ΔLERF to be between 1.0E-7 and 1.0E-6.

A.2. Internal Events LERF Model

No Large Early Release Frequency (LERF) F&Os are considered to be open. However, the following Supporting Requirements (SRs) have been affected and assessed as meeting Capability Category (CC) I: LE-B2, LE-C1, LE-C4, LE-D2, LE-D3, LE-F1, and LE-G3 (Finding-level F&O LE-E2-01 has been closed). Each of these SRs is discussed, as follows.

F&Os Regarding PRA Supporting Requirements	
SR ID:	**Peer Review CC Assessment:**
LE-B2 (note, no finding-level F&O)	Met CC I
Capability Category II Requirements:	
DETERMINE the containment challenges (e.g., temperature, pressure loads, debris impingement) resulting from contributors identified in LE-B1 using applicable generic or plant-specific analyses for significant containment challenges. USE conservative treatment or a combination of conservative and realistic treatment for non-significant containment challenges. If generic calculations are used in support of the assessment, JUSTIFY applicability to the plant being evaluated.	
Issue and Proposed Resolution:	
McGuire basically used a NUREG/CR-6595 model. They did address all "large-dry PWR containment," LERF contributors from Table 2-2.8.3 of the ASME/ANS PRA Standard with modifications as needed to address their ice condenser. However, McGuire did not explicitly determine the containment challenges in terms of pressures, temperatures and debris impingement for this analysis. Instead, they chose to use, via reference, a number of generic analyses, McGuire plant-specific analyses from the IPE, conservative assumptions to provide information on containment challenges and the basic NUREG/CR-6595 conclusions on the containment challenges from the various contributors. The approach used by McGuire is fully consistent with NUREG/CR-6595. The NRC has explicitly accepted the NUREG/CR-6595 approach as being sufficient for determination of LERF. **Possible Resolution:** To move from CC I to CC II, McGuire would need to provide more detail on the exact containment challenges in terms of pressures and temperatures for the significant contributors. The temperature and pressure information as well as a comparison to the McGuire containment ultimate pressures should be included in MCC-1535.00-00-0156 or MCC-1535.00-00-0051.	
Disposition of the Peer Review Finding:	
Steps taken to resolve Peer Review assessment: N/A. McGuire uses a LERF model based on the simplified LERF model in NUREG/CR-6595 [Reference 33]. While a NUREG/CR-6595 model is classified as Capability Category I, historically the NRC has indicated that a NUREG/CR-6595 model is of sufficient capability to support risk-informed applications.	

Subsequent to the 2012 peer review the LERF model was changed to be more consistent with this suggestion. This work performed was reviewed but is insufficient to change the SR CC: the SR is still met at CC I.

Evaluation of Peer Review assessment impact on proposed application:

The existing LERF analysis is sufficient for this ILRT Extension application. The LERF analysis uses the approach contained in NUREG/CR-6595 and relies on more generic analyses and conservative assumptions, instead of explicitly determining the containment challenges in terms of pressures, temperatures and debris impingement. The use of NUREG/CR-6595 is conservative and potentially overstates the LERF risk, thus total LERF is not subtracted from CDF when calculating delta LERF as this would lead to non-conservative results for this application. Instead, only the Class 2 and Class 8, which represent containment isolation failure and containment bypass (SGTR and ISLOCA), portions of LERF are subtracted from CDF when calculating delta LERF. Since containment isolation failure, SGTR, and ISLOCA represent containment failure modes that are not associated with accident progression, these failure modes are less likely to be conservative due to the simplified LERF analysis [Reference 33]. CC-1 is acceptable for this application.

F&Os Regarding PRA Supporting Requirements

SR ID:	Peer Review CC Assessment:
LE-C1 (note, no finding-level F&O)	Met CC I

Capability Category II Requirements:

DEVELOP accident sequences to a level of detail to account for the potential contributors identified in LE-B1 and analyzed in LE-B2. Compare the containment challenges analyzed in LE-B with the containment structural capability analyzed in LE-D and identify accident progressions that have the potential for a large early release. JUSTIFY any generic or plant-specific calculations or references used to categorized releases as non-LERF contributors based on release magnitude or timing. NUREG/CR-6595, App. A provides a discussion and examples of LERF source terms.

Issue and Proposed Resolution:

McGuire basically used a NUREG/CR-6595 model. They did address all "large-dry PWR containment," LERF contributors from Table 2-2.8.3 of the ASME/ANS PRA Standard with modifications as needed to address their ice condenser. However, McGuire did not explicitly determine the containment challenges in terms of pressures, temperatures and debris impingement for this analysis. Instead, they chose to use, via reference, a number of generic analyses, McGuire plant-specific analyses from the IPE, conservative assumptions to provide information on containment challenges and the basic NUREG/CR-6595 conclusions on the containment challenges from the various contributors. The approach used by McGuire is fully consistent with NUREG/CR-6595. The NRC has explicitly accepted the NUREG/CR-6595 approach as being sufficient for determination of LERF.

Possible Resolution:

To move from CC I to CC II, McGuire would need to provide more detail on the exact containment challenges in terms of pressures and temperatures for the significant contributors. The temperature and pressure information as well as a comparison to the McGuire containment ultimate pressures should be included in MCC-1535.00-00-0156 or MCC-1535.00-00-0051.

Disposition of the Peer Review Finding:

Steps taken to resolve Peer Review assessment:

N/A. McGuire uses a LERF model based on the simplified LERF model in NUREG/CR-6595 [Reference 33]. While a NUREG/CR-6595 model is classified as Capability Category I, historically the NRC has indicated that a NUREG/CR-6595 model is of sufficient capability to support risk-informed applications.

Subsequent to the 2012 peer review the LERF model was changed to be more consistent with this suggestion. This work performed was reviewed but is insufficient to change the SR CC: the SR is still met at CC I.

Evaluation of Peer Review assessment impact on proposed application:

The existing LERF analysis is sufficient for this ILRT Extension application. The LERF analysis uses the approach contained in NUREG/CR-6595 and relies on more generic analyses and conservative assumptions, instead of explicitly determining the containment challenges in terms of pressures, temperatures and debris impingement. The use of NUREG/CR-6595 is conservative and potentially overstates the LERF risk, thus total LERF is not subtracted from CDF when calculating delta LERF as this would lead to non-conservative results for this application. Instead, only the Class 2 and Class 8, which represent containment isolation failure and containment bypass (SGTR and ISLOCA), portions of LERF are subtracted from CDF when calculating delta LERF. Since containment isolation failure, SGTR, and ISLOCA represent containment failure modes that are not associated with accident progression, these failure modes are less likely to be conservative due to the simplified LERF analysis [Reference 33]. CC-1 is acceptable for this application.

F&Os Regarding PRA Supporting Requirements

SR ID:	Peer Review CC Assessment:
LE-C4 (note, no F&O)	Met CC I

Capability Category II Requirements:

INCLUDE model logic necessary to provide a realistic estimation of the significant accident progression sequences resulting in a large early release. INCLUDE mitigating actions by operating staff, effect of fission product scrubbing on radionuclide release, and expected beneficial failures in significant accident progression sequences. PROVIDE technical justification (by plant-specific or applicable generic calculations demonstrating the feasibility

of the actions, scrubbing mechanisms, or beneficial failures) supporting the inclusion of any of these features.

Issue and Proposed Resolution:

McGuire has a NUREG/CR-6595 LERF model. They updated the 6595 Containment Event Tree (CET) to reflect McGuire specifics. They have incorporated the top logic from the CET into their base model for ease of quantification and to directly capture dependencies from the internal initiator models. However, McGuire does not credit release scrubbing or the effects of beneficial failures. The basis is the assumption that these elements do not contribute significantly to reducing the LERF so excluding them to simplify the model is slightly conservative. McGuire meets CC I based on the use of the NUREG/CR-6595 model. The NRC has explicitly accepted the NUREG/CR-6595 approach as being sufficient for determination of LERF.

Possible Resolution:

Update LERF model to meet Capability Category II Requirements.

Disposition of the Peer Review Finding:

Steps taken to resolve Peer Review assessment:

N/A. McGuire uses a LERF model based on the simplified LERF model in NUREG/CR-6595 [Reference 33]. While a NUREG/CR-6595 model is classified as Capability Category I, historically the NRC has indicated that a NUREG/CR-6595 model is of sufficient capability to support risk-informed applications.

Evaluation of Peer Review assessment impact on proposed application:

The existing LERF analysis is sufficient for this ILRT Extension application. The LERF analysis uses the approach contained in NUREG/CR-6595 and relies on more generic analyses and conservative assumptions. The use of NUREG/CR-6595 is conservative and potentially overstates the LERF risk, thus total LERF is not subtracted from CDF when calculating delta LERF as this would lead to non-conservative results for this application. Instead, only the Class 2 and Class 8, which represent containment isolation failure and containment bypass (SGTR and ISLOCA), portions of LERF are subtracted from CDF when calculating delta LERF. Since containment isolation failure, SGTR, and ISLOCA represent containment failure modes that are not associated with accident progression, these failure modes are less likely to be conservative due to the simplified LERF analysis [Reference 33]. CC-1 is acceptable for this application.

F&Os Regarding PRA Supporting Requirements

SR ID:	Peer Review CC Assessment:
LE-D2 (note, no finding-level F&O)	Met CC I

Capability Category II Requirements:

EVALUATE the impact of containment seals, penetrations, hatches, drywell heads (BWRs), and vent pipe bellows and INCLUDE as potential containment challenges, as required. If generic analyses are used in support of the assessment, JUSTIFY applicability to the plant being evaluated.

Issue and Proposed Resolution:

McGuire has a NUREG/CR-6595 LERF model. McGuire chose to use the evaluation of the impact of accident progression conditions on containment seals, penetrations, hatches and vent pipe bellows embodied in NUREG/CR-6595.

Possible Resolution:

To move up from CC I to CC II, McGuire would need to perform a plant-specific evaluation of the impact of accident progression conditions on containment seals, penetrations, hatches, and vent pipe bellows and INCLUDE as potential containment challenges. This would involve evaluation of the performance of the seals and penetrations at several different containment temperatures consistent with the various accident progression sequences.

Disposition of the Peer Review Finding:

Steps taken to resolve Peer Review assessment:

N/A. McGuire uses a LERF model based on the simplified LERF model in NUREG/CR-6595 [Reference 33]. While a NUREG/CR-6595 model is classified as Capability Category I, historically the NRC has indicated that a NUREG/CR-6595 model is of sufficient capability to support risk-informed applications.

Subsequent to the 2012 peer review, the LERF notebook was revised to include the results of the 2013 self-assessment. The LERF notebook also provided rationales for meeting a number of SRs at CC I.

Evaluation of Peer Review assessment impact on proposed application:

The existing LERF analysis is sufficient for this ILRT Extension application. The LERF analysis uses the approach contained in NUREG/CR-6595 and relies on more generic analyses and conservative assumptions. The use of NUREG/CR-6595 is conservative and potentially overstates the LERF risk, thus total LERF is not subtracted from CDF when calculating delta LERF as this would lead to non-conservative results for this application. Instead, only the Class 2 and Class 8, which represent containment isolation failure and containment bypass (SGTR and ISLOCA), portions of LERF are subtracted from CDF when calculating delta LERF. Since containment isolation failure, SGTR, and ISLOCA represent containment failure modes that are not associated with accident progression, these failure modes are less likely to be conservative due to the simplified LERF analysis [Reference 33]. CC-1 is acceptable for this application.

F&Os Regarding PRA Supporting Requirements	
SR ID:	**Peer Review CC Assessment:**
LE-D3 (note, no finding-level F&O)	Met CC I

Capability Category II Requirements:

When containment failure location affects the event classification of the accident progression as a large early release, DEFINE failure location based on a realistic containment assessment that accounts for plant-specific features. If generic analyses are used in support of the assessment, JUSTIFY applicability to the plant being evaluated.

Issue and Proposed Resolution:

McGuire has a NUREG/CR-6595 LERF model. McGuire chose not to specifically identify the failure locations and chose to use the NUREG/CR-6595 simplification.

Possible Resolution:

To move up from CC I to CC II or better, McGuire would need to identify the likely containment failure locations for the severe accident sequences. The containment failure location analysis is available from the McGuire Containment structural analysis in the McGuire IPE Report (PRA Revision 1) Appendix G. This would need to be coupled with the individual sequences.

Disposition of the Peer Review Finding:

Steps taken to resolve Peer Review assessment:

N/A. McGuire uses a LERF model based on the simplified LERF model in NUREG/CR-6595 [Reference 33]. While a NUREG/CR-6595 model is classified as Capability Category I, historically the NRC has indicated that a NUREG/CR-6595 model is of sufficient capability to support risk-informed applications.

Subsequent to the 2012 peer review, the LERF notebook was revised to include the results of the 2013 self-assessment. The LERF notebook also provided rationales for meeting a number of SRs at CC I.

Evaluation of Peer Review assessment impact on proposed application:

The existing LERF analysis is sufficient for this ILRT Extension application. The LERF analysis uses the approach contained in NUREG/CR-6595 and relies on more generic analyses and conservative assumptions. The use of NUREG/CR-6595 is conservative and potentially overstates the LERF risk, thus total LERF is not subtracted from CDF when calculating delta LERF as this would lead to non-conservative results for this application. Instead, only the Class 2 and Class 8, which represent containment isolation failure and containment bypass (SGTR and ISLOCA), portions of LERF are subtracted from CDF when calculating delta LERF. Since containment isolation failure, SGTR, and ISLOCA represent containment failure modes that are not associated with accident progression, these failure

modes are less likely to be conservative due to the simplified LERF analysis [Reference 33]. CC-1 is acceptable for this application.

F&Os Regarding PRA Supporting Requirements	
SR ID:	**Peer Review CC Assessment:**
LE-E2 (note: Finding-level F&O LE-E2-01 has been closed) has been addressed to move from CC I to CC II.	Met CC II
Capability Category II Requirements:	
USE realistic parameter estimates to characterize accident progression phenomena for significant accident progression sequences resulting in a large early release. USE conservative or a combination of conservative and realistic estimates for non-significant accident sequences resulting in a large early release.	
Issue and Proposed Resolution:	
McGuire basically used the conservative parameter estimates from NUREG/CR-6595 to characterize the accident progression phenomena. This approach would satisfy CC I. However, Duke is using the Conditional Containment Failure Probabilities (CCFPs) from Revision 0 of NUREG/CR-6595 rather than the more restrictive values from Revision 1. To meet this requirement would require using the NUREG/CR-6595, Rev. 1 CCFP values or providing an engineering analysis to defend use of the older values. At the time of the peer review, Duke did have a white paper, "Conditional Containment Failure Probabilities for the McGuire and Catawba Large Early Release Frequency Models", November 2012, that discusses the basis for the use of the CCFPs from Rev. 0 of NUREG/CR-6595. However, this white paper was not provided as part of the official documentation for the review and as such, was not directly reviewed. A later review of this white paper indicates that Duke appears to have a reasonable basis for using the revision 0 CCFP values based on plant-specific analysis. Duke should include this information in their LERF analysis reports. **Possible Resolution:** To move from CC I to CC II would require using the NUREG/CR-6595, Revision 1 CCFP values or providing an engineering analysis to defend use of the older values.	
Disposition of the Peer Review Finding:	
Steps taken to resolve Peer Review assessment: McGuire uses a LERF model based on the simplified LERF model in NUREG/CR-6595 [Reference 33]. While a NUREG/CR-6595 model is classified as Capability Category I, historically the NRC has indicated that a NUREG/CR-6595 model is of sufficient capability to support risk-informed applications. Subsequent to the 2012 peer review the LERF model was changed to incorporate the Conditional Containment Failure Probability (CCFP) values used in NUREG/CR-6595,	

Revision 1. The work performed was reviewed and confirmed the use of the appropriate CCFP values. As a result, this effort is considered sufficient to disposition the finding.

This is now met at Capability Category II. A site-specific analysis was used to apply CCFP values to LERF cutsets. As documented in Reference 38, this update is resolved and incorporated in the model. Since this change has not been independently reviewed by a third party, this discussion is still reported.

Evaluation of Peer Review assessment impact on proposed application:

The existing LERF analysis is sufficient for this ILRT Extension application. A site-specific analysis was used to apply CCFP values to LERF cutsets. As documented in Reference 38, this update is resolved and incorporated in the model. This is now met at Capability Category II by Duke review and update. Even if independent review only considered this met at Capability Category I, the ILRT extension analysis was done assuming using the MNS LERF model is conservative and potentially overstates the LERF risk, thus total LERF is not subtracted from CDF when calculating delta LERF as this would lead to non-conservative results for this application. Instead, only the Class 2 and Class 8, which represent containment isolation failure and containment bypass (SGTR and ISLOCA), portions of LERF are subtracted from CDF when calculating delta LERF. Since containment isolation failure, SGTR, and ISLOCA represent containment failure modes that are not associated with accident progression, these failure modes are less likely to be conservative due to the simplified LERF analysis [Reference 33].

F&Os Regarding PRA Supporting Requirements	
SR ID:	**Peer Review CC Assessment:**
LE-F1 (note, finding-level F&O has been closed)	Met CC I
Capability Category II/III Requirements:	
PERFORM a quantitative evaluation of the relative contribution to LERF from plant damage states and significant LERF contributors from Table 2-2.8-3.	
Issue and Proposed Resolution:	
In [MCC-1535.00-00-051], McGuire documents the significant contributors to LERF in terms of contribution by initiating events. However, they did not document the relative contribution of contributors such as plant damage states, accident progression sequences, phenomena, containment challenges and containment failure modes. **Possible Resolution:** To move from CC I to CC II/III, McGuire needs to evaluate the relative contributions to LERF by such things as plant damage states, accident progression sequences, phenomena, containment challenges, and containment failure modes.	
Disposition of the Peer Review Finding:	
Steps taken to resolve Peer Review assessment:	

The most recent revision of the Model Integration Notebook documents the contribution to LERF by initiator, as well as by the most significant LERF sequences. Since McGuire uses a LERF model based on the simplified LERF model in NUREG/CR-6595 [Reference 33], and while a NUREG/CR-6595 model is classified as Capability Category I, historically the NRC has indicated that a NUREG/CR-6595 model is of sufficient capability to support risk-informed applications.

Subsequent to the 2012 peer review, the LERF notebook was revised to include the results of the 2013 self-assessment. The LERF notebook also provided rationales for meeting a number of SRs at CC I. The evaluation in the revised LERF notebook concluded that this SR was met CC I and that this was satisfactory. The notebook rationale for meeting CC I was reviewed. In addition, the results presented were reviewed and determined to include a high-level characterization by initiating event category, a list of significant LERF accident sequence cutsets, and lists of LERF risk-significant basic events.

Evaluation of Peer Review assessment impact on proposed application:

The existing LERF analysis is sufficient for this ILRT Extension application. The LERF analysis documents the significant contributors to LERF, by initiating events. However, the LERF analysis does not evaluate (and document) the relative contributors to LERF by such things as plant damage states, accident progression sequences, phenomena, containment challenges, and containment failure modes, required to meet CC II/III. The LERF analysis uses the approach contained in NUREG/CR-6595 and relies on more generic analyses and conservative assumptions. The use of NUREG/CR-6595 is conservative and potentially overstates the LERF risk, thus total LERF is not subtracted from CDF when calculating delta LERF as this would lead to non-conservative results for this application. Instead, only the Class 2 and Class 8, which represent containment isolation failure and containment bypass (SGTR and ISLOCA), portions of LERF are subtracted from CDF when calculating delta LERF. Since containment isolation failure, SGTR, and ISLOCA represent containment failure modes that are not associated with accident progression, these failure modes are less likely to be conservative due to the simplified LERF analysis [Reference 33]. CC-1 is acceptable for this application.

F&Os Regarding PRA Supporting Requirements	
SR ID:	**Peer Review CC Assessment:**
LE-G3 (note, finding-level F&O has been closed)	Met CC I
Capability Category II/III Requirements:	
DOCUMENT the relative contribution of contributors (i.e., plant damage states, accident progression sequences, phenomena, containment challenges, containment failure modes) to LERF.	
Issue and Proposed Resolution:	

In [MCC-1535.00-00-051], McGuire documents the significant contributors to LERF in terms of contribution by initiating events. However, they did not document the relative contribution of contributors such as plant damage states, accident progression sequences, phenomena, containment challenges and containment failure modes.

Possible Resolution:

To move from CC I to CC II/III, McGuire needs to evaluate the relative contributions to LERF by such things as plant damage states, accident progression sequences, phenomena, containment challenges, and containment failure modes.

Disposition of the Peer Review Finding:

Steps taken to resolve Peer Review assessment:

The most recent revision of the Model Integration Notebook documents the contribution to LERF by initiator, as well as by the most significant LERF sequences. Since McGuire uses a LERF model based on the simplified LERF model in NUREG/CR-6595 [Reference 33], and while a NUREG/CR-6595 model is classified as Capability Category I, historically the NRC has indicated that a NUREG/CR-6595 model is of sufficient capability to support risk-informed applications.

Subsequent to the 2012 peer review, the LERF notebook was revised to include the results of the 2013 self-assessment. The LERF notebook also provided rationales for meeting a number of SRs at CC I. The evaluation in the revised LERF notebook concluded that this SR was met CC I and that this was satisfactory. The notebook rationale for meeting CC I was reviewed. In addition, the results presented were reviewed and determined to include a high-level characterization by initiating event category, a list of significant LERF accident sequence cutsets, and lists of LERF risk-significant basic events.

Evaluation of Peer Review assessment impact on proposed application:

The existing LERF analysis is sufficient for this ILRT Extension application. The LERF analysis documents the significant contributors to LERF, by initiating events. However, the LERF analysis does not evaluate (and document) the relative contributors to LERF by such things as plant damage states, accident progression sequences, phenomena, containment challenges, and containment failure modes, required to meet CC II/III. The LERF analysis uses the approach contained in NUREG/CR-6595 and relies on more generic analyses and conservative assumptions. The use of NUREG/CR-6595 is conservative and potentially overstates the LERF risk, thus total LERF is not subtracted from CDF when calculating delta LERF as this would lead to non-conservative results for this application. Instead, only the Class 2 and Class 8, which represent containment isolation failure and containment bypass (SGTR and ISLOCA), portions of LERF are subtracted from CDF when calculating delta LERF. Since containment isolation failure, SGTR, and ISLOCA represent containment failure modes that are not associated with accident progression, these failure modes are less likely to be conservative due to the simplified LERF analysis [Reference 33]. CC-1 is acceptable for this application.

A.3. Internal Flooding

Two (2) internal Flooding PRA finding-level F&Os were considered to be open. Duke has resolved each of the finding-level F&Os, and they are discussed below [Reference 43]. Since these changes have not been independently reviewed by a third party, this discussion is still reported.

F&Os Regarding PRA Supporting Requirements		
F&O ID:	**Other Affected SRs:**	**Peer Review CC Assessment:**
IFSN A5-01	IFSN-A6, IFSN-A15, IFQU-A9	IFSN A5-01 Not Met (*IFSN-A6, IFSN-A15 & IFQU-A9 Met at CC I/II/III*)
Capability Category I/II/III Requirements:		
For each flood area not screened out using the requirements under other Internal Flood Supporting requirements (e.g. IFSO-A3 and IFSN-A12), IDENTIFY the SSCs located in each defined flood area and along flood propagation paths that are modeled in the internal events PRA model as being required to respond to an initiating event or whose failure would challenge normal plant operation, and are susceptible to flood. For each identified SSC, IDENTIFY, for the purpose of determining its susceptibly per IFSN-A6, its spatial location in the area and any flooding mitigative features (e.g., shielding, flood, or spray capability ratings).		
F&O Issue and Proposed Resolution:		
Although some of the flood areas appear to identify all PRA related SSCs in them, others do not. For example, the Unit 2 AFW Pump Room only lists the major pumps and the Safe Shutdown Panels. No additional valves or other PRA-related equipment located in the room are identified. Additionally, no spatial information of any of the PRA-related equipment in the flood areas appears to be documented in the Internal Flooding Analysis. Table 5-5 screening criteria may not be appropriate. **Basis for Significance:** SR IFSN-A5 not met. **Possible Resolution:** Using the list of PRA-related components contained in the Fire Database, develop a table for the Internal Flooding analysis that identifies the PRA-related equipment in each flood area, it spatial information, and any mitigative features available in the flood area. Once all SSCs in each identified flood area are identified, identify the susceptibility of each SSC in the flood area to flood-induced failure mechanisms (see IFSN-A6 requirements) and ensure they are included in the quantification process. Verify that the screening criteria for Table 5-5 is appropriate for all components.		

Disposition of the Peer Review Finding:
This F&O has been addressed and incorporated into the PRA model of record as follows. See Finding IFSN A17-01 for resolution of PRA equipment documentation. Assumption 2 in [MCC-153500-00-0123] provides justification as to why only limited equipment spatial information is documented. In all cases equipment was conservatively assumed to fail as a result of the different flooding mechanisms (as appropriate) and then refinement was performed as appropriate. Documentation includes spatial information as appropriate as refinement to the modeled scenarios were performed. For example, the CA pump room includes a detailed discussion of equipment in the room and its proximity to potential flooding hazards. Table 5-5 in CN-RAM-10-003 was compiled based on numerous flooding analyses and was cross-checked with McGuire plant staff to ensure Table 5-5 appropriately represented plant equipment susceptibility. This F&O has not been formally closed by a peer review team but is considered adequately addressed to not impact risk in a significant manner and met at CC I/II/II for IFSN-A5. **Evaluation of Peer Review assessment impact on proposed application:** Combined with the conservative analytical approach of failing equipment in a given flood scenario, the internal flooding analysis appropriately reflects the internal flood CDF and LERF and therefore the changes in CDF and LERF due to the extended ILRT interval. Therefore, no additional actions are needed to address this F&O for this application.

F&Os Regarding PRA Supporting Requirements		
F&O ID:	**Other Affected SRs:**	**Peer Review CC Assessment:**
IFSN A17-01	IFPP-A5, IFSO-A6, IFQU-A11	IFSN-A5; IFSO-A17; IFQU-A11 Not Met (*IFSO-A6 Met at CC I/II/III*)
Capability Category I/II/III Requirements:		
CONDUCT plant walk down(s) to verify the accuracy of information obtained from plant information sources and to obtain or verify a) SSCs located within each defined flood area b) flood/spray/other applicable mitigate features of the SSCs located within each defined flood area (e.g., drains, shields, etc.) c) pathways that could lead to transport to the flood area		
F&O Issue and Proposed Resolution:		
A review of the walk down forms provided in Appendix A of MCC-1535-121 identified that some critical information required to support the SRs mentioned above was not available on the walk down forms. For example, although the forms contained a field for Flood Sources,		

this appeared to only include tanks, and not piping in the area (note states that piping and corresponding lengths were determined from isometric and layout drawings), and the volume of the tanks was not always provided. Although there is a field on the form for PRA-related equipment in the Area, this information was not filled out consistently, resulting in some of the flood zones only listing major equipment such as pumps and panels, but not all PRA equipment in the room so it is not always possible to determine what is susceptible to flood-related impacts, including submergence and/or spray impacts in the room. Additionally, some of the information provided is contradictory for example, on the walk down for Unit 2 CA Pump Room although doors are listed, the Door Type and Door Sizes are given as N/A, and no information is provided with respect to which way the door opens. Some of the walk down forms said they assumed the information was the same as Unit 1 which implies that an actual walk down of the room was never performed.

Since the some of the original walk downs are incomplete with respect to identifying all the information required to satisfy the requirements of SRs IFSO-A6, IFPP-A5, IFSN-A17, IFQU-A11 (e.g., equipment locations in the rooms, door propagation pathways, some information on the forms is assumed information, pipe lengths and sizes are taken from isometric and layout drawings, etc.) walk downs to verify the information not documented during the original walk downs are required to ensure validity of the information.

Basis for Significance:

SR IFSN-A17 not met.

Possible Resolution:

Ensure that walk downs are performed and documented that specifically address the requirements of these SRs. If these walk downs were performed, ensure that the information is complete, and can be easily found.

Disposition of the Peer Review Finding:

This F&O has been addressed and incorporated into the PRA model of record as follows.

Walk downs have been verified and forms have been reviewed to ensure all fields have been filled out. The items identified by the peer review team have been addressed with the exception of more fully listing the PRA equipment located in a flood area. The walk down sheets continue to identify only the PRA-related equipment critical to developing the internal flooding PRA model sufficient to capture the impact of the flooding events. For example, a flood area may contain a motor-driven pump and associated motor operated suction and/or discharge valves. The impact on system operation is the same if either the motor-driven pump or motor operated valve is affected by a flooding event. It is therefore not necessary to list both the pump and valves to assess the flooding impact. Inclusion of either the pump or valve(s) is sufficient to assess the flooding impact on PRA-related equipment in the flood area. Thus, the level of detail presented in the walk down forms is adequate and a statement justifying the level of detail presented has been incorporated into the documentation.

This F&O has not been formally closed by a peer review team but is considered adequately addressed to not impact risk in a significant manner and met at CC I/II/II for IFSN-A17 and IFQU-A11.

Evaluation of Peer Review assessment impact on proposed application:

The actions above resolve the issue and therefore, no additional actions are needed to address this F&O for this application. There is no impact on the ILRT extension application because the finding is documentation related.

A.4. High Winds

Eight (8) High Winds PRA finding-level F&Os are considered to be open. Each of the open finding-level F&Os is discussed, as follows.

F&Os Regarding PRA Supporting Requirements		
F&O ID:	**Other Affected SRs:**	**Peer Review CC Assessment:**
WHA-A2-2		Met at CC II/III
Capability Category II/III Requirements:		
In evaluating the hazard from hurricanes, USE the state-of-the-art hurricane hazard analysis methodology and up-to-date databases on hurricane occurrences, intensities, etc. PROPAGATE uncertainties in the models and parameter values in order to obtain a family of hazard curves from which a mean hazard curve can be derived.		
F&O Issue and Proposed Resolution:		
Hurricane missiles and straight wind produced missiles were not separately analyzed; their risks were assumed to be covered by the results of tornado missile risk analysis. **Basis for Significance:** This may be somewhat conservative and assumed to be accounted in the uncertainty analysis. Vol. III, Section 5.1 provides some arguments for not treating the straight wind missiles separately. **Possible Resolution:** Further justification (may be qualitative) for not treating the hurricane generated missiles should be provided.		
Disposition of the Peer Review Finding:		
Steps taken to resolve Peer Review assessment: The discussion of tornado and straight line wind missiles in Volume III, Section 5.1 has been revised to include hurricane wind missiles within the same context as straight-wind missiles.		

In addition, a new consideration has been added, regarding the potential conservatism of using of the tornado missile simulation results for straight wind and hurricane missiles.

Evaluation of Peer Review assessment impact on proposed application:

The conservatism regarding hurricane missiles results in a larger delta LERF against acceptance criteria of RG 1.174. The hurricane risk at MNS is a small portion of the high wind risk and thus the conservative use of hurricane missile does not change the outcome of the acceptability of this application.

F&Os Regarding PRA Supporting Requirements

F&O ID:	Other Affected SRs:	Peer Review CC Assessment:
WFR-A1-1		Met at CC II/III

Capability Category II/III Requirements:

In evaluating wind fragilities of structures and components (e.g., tanks, transformers, diesel-generator exhaust stack, piping, and intake pumps), USE plant-specific data. In the assessment, INCLUDE non-safety structures that could fall into / onto safety-related structures, thereby causing damage. In this evaluation, INCLUDE the findings of a plant walkdown.

F&O Issue and Proposed Resolution:

Calculation MCC-1535.00-00-0178, High Wind Hazard and Fragility Analysis of MNS, Volume I, states (page 48) that the Unit 1 CAST is a vented tank and, as such, is not vulnerable to failure from atmospheric pressure change (APC) loading. However, since the tank is essentially full, venting the tank may not be adequate to relieve the differential pressure. The basis for screening this tank should be based on the capacity of the tank to withstand the additional differential pressure, not because it is vented.

Basis for Significance:

The decrease in atmospheric pressure, when added to the hydrostatic pressure due to water in the tank, may exceed design stresses and lead to failure of the tank.

Possible Resolution:

Verify the tank can withstand the additional stresses from the APC. Review the model for other tanks that also may be screened because they were vented and verify they are appropriately screened or evaluated for the additional stress.

Disposition of the Peer Review Finding:

Steps taken to resolve Peer Review assessment:

The original screening of the Unit 1 CAST (as well as the Unit 1 CA CST, and FWST) for APC loading because the tanks were vented was not correct. However, subsequent calculations have been performed and discussed in Volume I to analyze stresses within the tanks and confirm that they can be screened for APC loading in the fragility analysis.

Specifically, Revision 1 of Volume I, Attachment 3 provides details on the screening of these 3 tanks for APC loading. The revised screening calculates the hoop stress within the shell of each tank assuming that the tank is full of water with and without the MNS 3 psi APC loading prescribed in the MNS UFSAR and comparing the hoop stress in the tank to the stress levels allowed by the code the tanks were built to.

All 3 tanks were designed using the AWWA D100 code per UFSAR Table 3-4. This code specifies a maximum allowable stress of 15000 psi in the tank shell. Maximum hoop stresses including the pressure from a full tank of water and the APC loading, acting in the same direction, for each tank was found to be:

- U1 CAST – 14,480 psi
- U1 FWST – 13,572 psi
- U1 CA CST – 6,770 psi

Since all of these stresses are within the allowable limits of the design code, each of these tanks is screened for APC loading.

References to tanks being screened for APC based on venting in Volume I Sections 2, 8.5 (CA CAST), and 8.8 (CAST and FWST) have been removed and replaced with a discussion of the screening calculations now provided in Volume I, Attachment 3.

Evaluation of Peer Review assessment impact on proposed application:

As discussed above, appropriate design information, including equipment specifications, has been used to evaluate the APC loading of the various tanks. Since this analysis did not change any input of the High Winds PRA model, there is no impact on this application.

F&Os Regarding PRA Supporting Requirements		
F&O ID:	**Other Affected SRs:**	**Peer Review CC Assessment:**
WPR-A3-1		Met at CC I-III
Capability Category I/II/III Requirements:		
ENSURE that the PRA systems models reflect wind-caused failures as well as other unavailabilities and human errors that give rise to significant accident sequences or significant accident progression sequences.		
F&O Issue and Proposed Resolution:		

Main Steam and Main Feedwater lines are adjacent to the Emergency Diesel Generator (EDG) Intakes. A failure of one of these high energy lines (e.g., due to a high winds generated missile) could create a steam environment which could potentially disable both EDGs. Such impact is not modeled, nor is an evaluation of the postulated impact evaluated.

Basis for Significance:

During high wind scenarios, which often result in a Loss of Offsite Power (LOOP), the availability of the EDGs has greater importance. A high wind scenario that results in the loss of the EDGs, as well as a LOOP, will (if the SSF is not available due to the high wind scenario) result in core damage (reactor coolant pump seal LOCA without injection).

Possible Resolution:

Evaluate the above scenario for feasibility and frequency. The analysis should look at various timing combinations (e.g., missile puncturing high energy line first, then LOOP; LOOP and then missile damage to high energy line).

Disposition of the Peer Review Finding:

Steps taken to resolve Peer Review assessment:

This Finding was addressed in multiple volumes by ARA and by Hughes. ARA revised the screening of the Diesel Generator rooms and the wind pressure fragilities of the Main Steam and Main Feedwater lines between the Turbine Building and the Main Steam doghouses.

Structural interaction failure modes for wind pressure and missile failure of the Main Steam and Main Feedwater lines were analyzed for all equipment inside the MNS Unit 1 Diesel Generator rooms. Because Main Steam and Feedwater pipes run just above the Diesel Generator Building and directly past the generator air intakes, any failure of these lines due to wind pressure or wind missile that leads to a line break will send steam directly into the generator air intakes. Therefore, failure of the Main Steam and Feedwater lines due to wind pressure or missile is modeled as also failing both diesel generators.

Initial propagation of this screening change through the PRA model showed the wind pressure fragility of the Main Steam and Feedwater lines as a major contributor to the plant CDF. Because of this, the wind pressure fragilities of these lines were re-evaluated based on additional calculations provided by Duke Energy. The re-evaluation provided a significantly higher wind loading, (211mph vs 143 mph for Main Feedwater, and 339 mph vs 143 mph for Main Steam lines). Incorporating this re-evaluation, along with adding separate failure modes for wind pressure and missiles into the PRA, reduced the calculated CDF / LERF risk.

Evaluation of Peer Review assessment impact on proposed application:

The changes made in the High Winds PRA for wind loading and missile impacts on piping near the EDG was explicitly included in the High Winds PRA and thus no additional changes are required for this application.

F&Os Regarding PRA Supporting Requirements		
F&O ID:	**Other Affected SRs:**	**Peer Review CC Assessment:**
WPR-A3-2		Met at CC I-III

Capability Category I/II/III Requirements:

ENSURE that the PRA systems models reflect wind-caused failures as well as other unavailabilities and human errors that give rise to significant accident sequences or significant accident progression sequences.

F&O Issue and Proposed Resolution:

The recirculation line from the High Pressure Injection pumps goes to the FWST, which is outside. A missile that crimps this line so recirculation flow is severely restricted could fail both NI (Safety Injection) pumps. Such impact is not modeled.

Basis for Significance:

Loss of injection during a loss of offsite power event can be a contributor to core damage.

Possible Resolution:

Evaluate the above scenario for feasibility and frequency.

Disposition of the Peer Review Finding:

Steps taken to resolve Peer Review assessment:

A sensitivity analysis was performed by rerunning the MNS TORMIS model with the FWST recirculation pipe added as a part of the FWST target group. This sensitivity analysis showed that the effect of adding the recirculation pipe to the analysis was minimal (0-4% to the fragility for each wind speed). As a result, it was concluded that revision of the wind missile fragility of the FWST reported in Rev 0 is not warranted.

Evaluation of Peer Review assessment impact on proposed application:

As discussed above, the CDF and LERF impact of the potential crimping of the HPI recirculation line is negligible. Additionally, the set of fragilities for the FWST contribute < 1% to HW CDF. Therefore, this has no impact on the acceptability of this application.

F&Os Regarding PRA Supporting Requirements

F&O ID:	Other Affected SRs:	Peer Review CC Assessment:
WPR-A5-1		Met at CC I-III

Capability Category I/II/III Requirements:

In the human reliability analysis (HRA) aspect, EVALUATE additional stresses that can increase the likelihood of human errors or inattention, compared to the likelihood assigned in the internal events HRA when the same activities are undertaken in non-high-wind-event accident sequences. Whether or not increases in error probabilities are used, JUSTIFY the basis for this decision about what error rates to use.

F&O Issue and Proposed Resolution:

Various places in the report (e.g., Section F.4.2) cite procedural guidance in RP/0/B/5700/027 that:

- operators will reduce loads or begin shutdown (4 hours) prior to anticipating high winds (hurricanes) arriving onsite,
- operators will have reviewed anticipated procedures,
- operators will be aware of equipment that can potentially be impacted by the storm or LOOP,
- additional staff would be available.

Section F.4.2 (and other places) states the plant is 'required' to be placed in hot standby well before the full strength of the storm is encountered.

This procedure citation is used to justify no increase in Performance Shaping Factors (stress, multiple procedures, and workload) during a hurricane, so HFEs during a hurricane are no different than during the internal event model.

In actuality, the procedure only directs 'evaluating' the performance of these activities. Therefore, it is possible that these actions would be evaluated and NOT performed, so crediting these actions as the basis for not impacting the PSFs is inadequate.

Basis for Significance:

Failure to evaluate the impact of high winds on human actions and on the dependency of such actions could lead to inaccurate PRA results and incorrect insights.

Possible Resolution:

Re-word the citation of the above procedure to eliminate reference that the actions would definitely be performed.

Disposition of the Peer Review Finding:

Steps taken to resolve Peer Review assessment:

The High Winds analysis has been updated by removing text that refers to a requirement of placing the plant in hot standby, and providing justification for the applied PSF in the HRA analysis. The text now states "Per procedural guidance in RP/0/B/5700/027 on High Winds or Hurricane Preparation [2], operators evaluate conditions to determine if it is necessary to reduce loads or begin shutdown (4 hours) prior to anticipating high winds arriving onsite. In anticipation of the high wind, operators will have reviewed anticipated procedures, operators will be aware of equipment that can potentially be impacted by the storm or LOOP and additional staff would be available. This procedure for a hurricane threat directs mitigation measures such as increased plant staffing, site preparation, staging of key personnel, procedure review, and protection of exposed sensitive equipment. The plant will likely be placed in hot standby well before the full strength of the storm is encountered...."

Evaluation of Peer Review assessment impact on proposed application:

The HRA PSF in the High Winds PRA are appropriate for the various scenarios, given the preparation time, procedural review, and potentially added staffing. Thus there is no impact on the ILRT extension.

F&Os Regarding PRA Supporting Requirements

F&O ID:	Other Affected SRs:	Peer Review CC Assessment:
WPR-A6-1		Met at CC I-III

Capability Category I/II/III Requirements:

If any screening is performed, PERFORM it using defined criteria that are documented in the PRA.

F&O Issue and Proposed Resolution:

In Table F-5.1, Screening of Post-Initiator Actions Based on High Winds Applicability, Human Failure Event (HFE) SX00LPRDHE (Operators Fail to Cooldown and Depressurize to LPR Conditions) was screened from review for impact by high winds, stating that the action was set to 1.0 in the internal events model. However, in section 7.5.4.1 (Operator Action Importance (CDF)), Table 7-32 lists Operator (combination) Action ZHFC-2-104 with a RAW of 1.53 (combination of SX00LPRDHE and TRECIRCDHE). This combination event does not show up in Table F-10.1 or Table F-10.2. Event TRECIRCDHE has a non-tornado value of 4.5E-3. In the cutsets, this combination event has a value of 4.0E-4. It appears HFE SX00LPRDHE was inappropriately screened from evaluation of impact by high winds and the combination event was not addressed for dependency. This may apply to other HFEs where the screening indicates a value of 1.0 in the internal events model.

Basis for Significance:

Failure to evaluate the impact of high winds on human actions and on the dependency of such actions could lead to inaccurate PRA results and incorrect insights.

Possible Resolution:

Evaluate SX00LPRDHE for impact by high winds and evaluate the dependency of combination event ZHFC-2-104 for impact by high winds. Review other HFEs that were screened because they were assigned a value of 1.0 in the internal events model to verify they were not inappropriately screened.

Disposition of the Peer Review Finding:

Steps taken to resolve Peer Review assessment:

Review of HFEs that were screened because they were assigned a value of 1.0 in the internal events was conducted. It was determined that four (4) HFEs were inappropriately set to 1.0: LLPIOPSDHE, SAGRCL1DHE, SX00LPRDHE and SX00RHRDHE. Therefore, these HFEs were retained for detailed analysis. In all cases, it was determined that due to the length of time available for the action to be completed, additional adjustments due to high wind initiators was not warranted. The high wind-adjusted HFEs used the same HEP as the IE HFEs. In turn, the dependency analysis was reviewed combinations involving the additional HFEs were included (e.g., ZHFC-2-104 and ZHFC-3-060). These dependent combinations were evaluated for high wind effects.

Evaluation of Peer Review assessment impact on proposed application:

The review of HFEs and subsequent use of detailed analysis is directly incorporated into the HW PRA model used in the analysis for the ILRT extension and the associated delta LERF metric. This direct impact in the analysis resolves concerns with the F&O.

F&Os Regarding PRA Supporting Requirements		
F&O ID:	**Other Affected SRs:**	**Peer Review CC Assessment:**
WPR-C1-1		Met at CC I-III
Capability Category I/II/III Requirements:		
DOCUMENT the wind plant response analysis and quantification in a manner that facilitates PRA applications, upgrades and peer review.		
F&O Issue and Proposed Resolution:		
There is no summary or conclusions provided in the summary document. Section 8.0 "Conclusions and Recommendations" provides no real conclusions or recommendations. To obtain significant insights and conclusion requires tracing through other parts of the document or other reference documents. Significant initiating contributors or component failures are not easily discerned. **Basis for Significance:** Supporting requirement WPR-C1 states, "Document the wind plant response analysis and quantification in a manner that facilitates PRA applications, upgrades and peer review." Although the results and insights could be obtained, this required examining several documents and performing some manipulation to arrive at such information as major initiating events contributing to CDF. **Possible Resolution:** Update Section 8.0 to summarize significant initiators (e.g., TB siding causes LOOP), CDF contributors (e.g., EDG failure to run) and any recommendations for plant improvements to mitigate high winds and tornadoes.		
Disposition of the Peer Review Finding:		
Steps taken to resolve Peer Review assessment: A new subsection was added to the high winds analysis summarizing significant findings of the HWPRA including recommendations to mitigate high-wind hazards. **Evaluation of Peer Review assessment impact on proposed application:** This F&O addresses needed improvements in the discussion of the results of the HWPRA. Thus, this F&O does not directly impact the HW CDF or LERF results used in the analysis for the risk impact of the ILRT extension application.		

F&Os Regarding PRA Supporting Requirements

F&O ID:	Other Affected SRs:	Peer Review CC Assessment:
WPR-C1-2		Met at CC I-III

Capability Category I/II/III Requirements:

DOCUMENT the wind plant response analysis and quantification in a manner that facilitates PRA applications, upgrades and peer review.

F&O Issue and Proposed Resolution:

Summary document MCC-1535.00-00-0178, MNS High Wind Probabilistic Risk Assessment (HWPA) has numerous errors that need correction, especially in citations of References and Assumptions: e.g.,

1. Page 17, 4.1, Assumption 16, says Reference 3.3-6, Step 3.4, directs the operators to trip the reactor with sustained (>15 minutes) High Winds > 95 mph. That procedure step does not direct that activity. The Assumption should refer to Reference 3.3-7, Step 3.4.
2. Page 26, next to last paragraph, states "Assumption #3 states that all high-wind events of F2 ...". This should be Assumption #4.
3. Table 7-5, 7-6, and 7-7 indicate the HW Interval Mid-point (mph) for 260 mph HW speed is 259.0. This should say 289 mph (see Vol IIB, Fig 8-16, Table 8-7).
4. F.5.1 says the EPRI guidance is reference 6 – it is actually reference 9. Also in line before F.7.1, in F.7.4, in Table F-8.2 Note, and in Table F-8.3 Note.

Numerous other examples exist.

Basis for Significance:

Inaccurate citations and errors hinder PRA applications, upgrades, and peer review.

Possible Resolution:

Review the document and ensure References and Assumptions match the citations in the document. Correct errors.

Disposition of the Peer Review Finding:

Steps taken to resolve Peer Review assessment:

Errors in documentation sited in this F&O were corrected including:

1. Updated reference in Assumption #16 from Section 4.1 of MCC-1535-00-00-0178.
2. Updated correct assumption number from Section 7.2.2.4 of MCC-1535-00-00-0178.
3. Updated F5 HW Interval Mid-point to 289mph from Section 7.3.3.1 of MCC-1535-00-00-0178.
4. Reference to EPRI Guidance changed to [9] in all cases. Corrections made in sections F.5.1, F.6, F.7, F.7.4, F.7.7, F.8, Table F-8.2 Note, and in Table F-8.3 Note all in from Section 4.1 of MCC-1535-00-00-0178 Attachment F.

Evaluation of Peer Review assessment impact on proposed application:

This F&O deals with inconsistency in the documentation of references and assumption citations. These were resolved in the HWPRA documentation, but have no impact on the High Winds CDF or LERF results or the calculation of the delta LERF for this application.

A.5. Fire

Twenty-two (22) Fire PRA finding-level F&Os are considered to be open. Additionally, 12 Supporting Requirements (SRs) have been assessed as meeting Capability Category (CC) I, four (4) of which have finding-level F&Os written against them. Each of these finding-level F&Os, as well as the SRs assessed as meeting CC-I, are discussed, as follows

F&Os Regarding PRA Supporting Requirements		
F&O ID:	**Other Affected SRs:**	**Peer Review CC Assessment:**
PP-B2-01		Not Met
Capability Category II/III Requirements:		
If partitioning credits wall, ceiling, or floor elements that lack a fire-resistance rating, JUSTIFY the judgment that the credited element will substantially contain the damaging effects of fires given the nature of the fire sources present in each compartment separated by the nonrated partitioning element.		
F&O Issue and Proposed Resolution:		
Drawing MC-1384-07.17-00 (draft) Note (1) states that six HVAC penetrations through the reactor building wall do not contain fire dampers - there is no mention of this in the partitioning documentation or why this is acceptable. Fire areas 25 to 32 and/or 33. Additional examples: Notes in drawings 17-00, 15-00, 14-00, and 13-00 describe multiple floors/ceiling between fire areas 29 (767&750), 28(767&750), 25(767), & 21 (750), 14 (733), and 4 (716) that are not fire stopped/ fire sealed, but these are listed as different fire areas whereas fire area 13 spans elevations 716 and 733 due to lack of fire stopping between elevations. SR requires that Justification be provided for elements that lack fire resistance rating. **Basis for Significance:** SR requires justification. **Possible Resolution:** Include justification or evaluation for FAs or combine areas.		
Disposition of the Peer Review Finding:		
Steps taken to resolve Peer Review assessment: While the Fire PRA did not formally disposition the fire protection drawing notes, the fire scenario walk downs would evaluate non-rated barriers that are relied on to truncate the zone of influence and identify targets in exposed compartments from an adjacent exposing compartment. The six non-rated HVAC penetrations through the reactor building wall (per drawing MC-1384-07.17-00) were evaluated to have no impact on the Fire PRA. While this evaluation was not formally documented, it is believed to meet the intent of the F&O.		

Additionally, unsealed wall between the Service Building and the Turbine Buildings and the non-committed firewalls separating the Essential Switchgear Rooms from the adjacent Penetration Rooms that were credited in the Fire PRA were addressed during the Fire Risk Evaluation calculation. The open communication between FA 4 and FA 14 and between FA 14 and FA 21 via the open stairwells was also evaluated to have no impact on the Fire PRA since there are no in-situ combustibles or fixed ignition sources which would contribute to fire propagating across the boundary.

Evaluation of Peer Review assessment impact on proposed application:

The evaluations performed of the non-rated barriers, as described above, are sufficient to analyze this application. A lack of formal documentation will not impact the Fire PRA model used in the analysis of the ILRT extension application or delta LERF. Therefore, the above completed actions address this assessment for this application.

F&Os Regarding PRA Supporting Requirements

F&O ID:	Other Affected SRs:	Peer Review CC Assessment:
PP-B7-01		Not Met

Capability Category I/II/III Requirements:

Conduct a confirmatory walk down of locations within the global analysis boundary to confirm the conditions and characteristics of credited partitioning elements.

F&O Issue and Proposed Resolution:

Calculation for Plant Partitioning and Ignition Frequency Calculation does not describe any walk down elements completed to meet the requirements of this SR. The Self-Assessment lists that these will be completed in later confirmatory walk down but the loop is not closed in this documentation that these walk downs were completed or addressed the elements of the partitioning SR. Discussion with plant analyst determined that additional walk down information related to plant partitioning was not confirmed or documented since the plant partitioning is based upon room barriers.

Basis for Significance:

Nothing in provided documentation to address SR.

Possible Resolution:

Document or provide proof in any other way that a walk down was performed to meet the requirements of PP-B7, including the condition and characteristics of partitioning elements.

Disposition of the Peer Review Finding:

Steps taken to resolve Peer Review assessment:

A subsequent walk down was conducted for plant partitioning and is documented in the Fire Scenario Report. This walk down addressed plant partitioning.

Evaluation of Peer Review assessment impact on proposed application:

The Fire PRA CDF and LERF is adequate for this application and to determine delta LERF. Therefore, no additional actions need be performed to address this assessment for this application.

F&Os Regarding PRA Supporting Requirements

F&O ID:	Other Affected SRs:	Peer Review CC Assessment:
PP-C3-01		Not Met

Capability Category I/II/III Requirements:

DOCUMENT the general nature and key or unique features of the partitioning elements that define each physical analysis unit defined in plant partitioning in a manner that facilitates Fire PRA applications, upgrades, and peer review.

F&O Issue and Proposed Resolution:

Description of the general nature and key or unique features of the partitioning elements is limited to a single statement that plant fire areas/rooms are used as the plant partitions. This is not sufficient documentation of these elements as fire barriers as noted in F&O PP-B2-01.

Basis for Significance:

Documentation does not meet the requirement to describe elements of the partitioning.

Possible Resolution:

Provide the FHA or similar justification for room fire barriers.

Disposition of the Peer Review Finding:

Steps taken to resolve Peer Review assessment:

A subsequent walk down was conducted for plant partitioning and is documented in the Fire Scenario Report. Fire Area boundaries are described in the Fire Protection Design Basis Specification. No non-conforming conditions were noted during the Fire PRA walk downs.

Evaluation of Peer Review assessment impact on proposed application:

The Fire PRA CDF and LERF is adequate for this application and to determine delta LERF. Therefore, no additional actions need be performed to address this assessment for this application.

F&Os Regarding PRA Supporting Requirements		
SR ID:	**Other Affected SRs:**	**Peer Review CC Assessment:**
CS-C1 (note, no F&O)		Not Met
Capability Category I/II/III Requirements:		
DOCUMENT the cable selection and location methodology applied in the Fire PRA in a manner that facilitates Fire PRA applications, upgrades, and peer review.		
Summary of Assessment:		
The Cable Selection documentation is well organized, but requires more detail in order to facilitate Fire PRA applications, upgrades and peer review. Therefore, the SR is judged to be not met.		
Disposition of the Peer Review Assessment:		
As stated in the Summary of Review Results, in the peer review report: The MNS cable database ARTRAK is an excellent source of information for the FPRA in general, and Cable Selection task in particular. The ARTRAK has been noted as an item of Strength. The Cable Selection SRs were essentially met. The routing has been assumed for certain cables and it appears reasonable, but a need for better documentation of the basis for the assumed routing has been identified. There is need for better documentation of the Y2 components, i.e., the components not in the Appendix R analysis but modeled in the fire PRA plant response model. **Steps taken to resolve Peer Review assessment:** Enhanced details on Y1 and Y2 cables were added to section 2 of the Cable Selection Report in order to facilitate Fire PRA applications. The report now states that Y1 and Y2 cables are comprised of thermoset cables constructed with flame retardant cross-linked polyethylene insulation, an interlocking armor and a PVC exterior jacket. **Evaluation of Peer Review assessment impact on proposed application:** Documentation enhancements that have been completed are sufficient to perform an analysis on the ILRT extension application; therefore, no additional actions need to be performed to address this assessment for this application.		

F&Os Regarding PRA Supporting Requirements		
F&O ID:	**Other Affected SRs:**	**Peer Review CC Assessment:**
CS-C3-01		Not Met

Capability Category I/II/III Requirements:

If the provision of SR CS-A11 is used, DOCUMENT the assumed cable routing and the basis for concluding that the routing is reasonable in a manner that facilitates Fire PRA applications, upgrades, and peer review.

F&O Issue and Proposed Resolution:

The assumed routing appears reasonable, but a clearer basis for the routing needs to be provided. This SR is judged to be met, but a finding is made to provide more detailed documentation. Although the assumed cable routing was judged to be reasonable, the documentation of this assumed routing was found to not be sufficient to facilitate Fire PRA applications, upgrades and peer review. This is a documentation-related F&O and is judged to be of sufficient importance to be classified as a finding. Complete documentation is required to meet this SR.

Basis for Significance:

The requirements of this SR appear to have been met, but more documentation is needed to facilitate review and PRA applications. Quantitative impact to the base PRA is not overly significant: 35% reduction if all assumed routing is assumed to be available and a 56% increase if no assumed routing is credited.

Any assumed cable routing however, may be important for PRA applications, and therefore a finding is made.

Possible Resolution:

Document a clearer basis for assumed routing. For example, "13, 17A, 18A, 19, 20, 24, TB1, TB2, SRV" appears to be a reasonable routing; however, this is not readily apparent based on the documentation.

Disposition of the Peer Review Finding:

Steps taken to resolve Peer Review assessment:

A clearer basis for the Y3 cable routing was included in section 2.3 and Appendix B of the Cable Selection Calculation. Additionally, cable selection has since been expanded to address numerous Y3 components; consequently, there are significantly fewer Y3 components. The impact of the Y3 components on quantification is relatively minimal as discussed in the sensitivity analysis within the Fire PRA Summary Report. Engineering judgment is a reasonable alternative to cable routing of Fire PRA components of lesser importance. Cable routing would not necessarily eliminate the assumed failure in those

scenarios where failure is assumed nor would it necessarily result in additional failures where the Y3 component is credited.

The typical information in Appendix B has been supplemented by considering selected cables (from drawings) and associated routes from the cable & raceway database as the basis for the credit by exclusion.

Evaluation of Peer Review assessment impact on proposed application:

Documentation enhancements that have been completed are sufficient to perform an analysis on the ILRT extension application; therefore, no additional actions need to be performed to address this assessment for this application.

F&Os Regarding PRA Supporting Requirements

F&O ID:	Other Affected SRs:	Peer Review CC Assessment:
CS-C4-01		Not Met

Capability Category I/II/III Requirements:

DOCUMENT the review of the electrical distribution system overcurrent coordination and protection analysis in a manner that facilitates Fire PRA application, upgrades, and peer review.

F&O Issue and Proposed Resolution:

The electrical distribution system overcurrent coordination and protection analysis does not yet include breaker coordination evaluations of the additional power supplies evaluated for the fire PRA (power supplies associated with Y2 components, including the evaluation of offsite power availability). Any additional circuits and cables whose failure could challenge power supply availability due to inadequate electrical overcurrent protective device coordination have already been identified and addressed by the fire PRA model. This finding is made to complete the overcurrent coordination and protection analysis documentation.

Based on discussions with cable selection analysts (AREVA), proper overcurrent coordination and protection with respect to common enclosure considerations (fire ignited in a secondary location due to excessive ohmic heating) have been addressed based on the underlying assumption that the plant design basis satisfies all applicable electrical code requirements. A limited scope review was also performed of plant design changes that involved changes to electrical protective devices to ensure compliance to applicable electrical codes was followed. This review is ongoing. The basis for this assumption and the details of the limited scope review have not yet been formally documented, and therefore a finding has been made.

Basis for Significance:

Complete documentation is required to meet this SR.

Possible Resolution:

Include the analysis of electrical distribution system overcurrent coordination and protection for Y2 components in the Breaker Coordination calculation, as well as documentation of common enclosure considerations.

Disposition of the Peer Review Finding:

Steps taken to resolve Peer Review assessment:
Results from Section 6.0 of the McGuire Appendix R Coordination Study have been included in the Fire PRA within the cable footprint for the affected components as discussed in Section 2.1.2 of the Cable Selection Report. The required power sources for the Y2 components have since been evaluated for breaker coordination concerns.

Evaluation of Peer Review assessment impact on proposed application:

Documentation enhancements that have been completed are sufficient to perform an analysis for the ILRT extension. Therefore, no additional actions need to be performed to address this assessment for this application.

F&Os Regarding PRA Supporting Requirements

F&O ID:	Other Affected SRs:	Peer Review CC Assessment:
PRM-B2-01		Not Met

Capability Category I/II/III Requirements:

VERIFY the peer review exceptions and deficiencies for the Internal Events PRA are dispositioned, and the disposition does not adversely affect the development of the Fire PRA plant response model.

F&O Issue and Proposed Resolution:

It is not clear from the fire PRA documentation that peer review exceptions and deficiencies will not adversely affect the development of the Fire PRA plant response model.

Basis for Significance:

There is a potential significant impact on the fire PRA plant response model.

Possible Resolution:

Resolve peer review exceptions and deficiencies of the internal events PRA or demonstrate more clearly that they have no adverse effects to the fire PRA plant response model. For example, the modeling of pre-initiator HFEs and updates to basic event data, accounting for running/standby components, may significantly impact the risk profile, and resolution would therefore be needed.

Disposition of the Peer Review Finding:

Steps taken to resolve Peer Review assessment:

The MNS IEPRA models pre-initiator HEPs, but it does not model pre-initiator miscalibration events. An internal events sensitivity study was performed in support of the RI-ISI application which added four miscalibration events, including miscalibration of the FWST level instrumentation, to the internal events model based on an ongoing update to the MNS internal events PRA. None of the events occurred in cutsets above the truncation limit with the internal events PRA initiators. This sensitivity study provides support for the judgment that the absence of the miscalibration errors is not significant to the Fire PRA results.

Additionally, when compared to post-initiator HEPs and fire-induced failures, miscalibration HEPs and other pre-initiator HEPs are not expected to significantly contribute to overall equipment unavailability.

Analysis of data trends documented in NUREG/CR-6928 indicates that there are no statistically significant trends in generic component reliability data over periods of up to ten years. Consequently, the conversion to a more recent generic database would not be expected to have significant impact on the Fire PRA results.

The MNS Applications calculation was updated to include the above discussion.

Evaluation of Peer Review assessment impact on proposed application:

Pre-initiator HEPs have been evaluated, and are not expected to significantly contribute to overall equipment unavailability. Analysis of data trends documented in NUREG/CR-6928 indicates that there are no statistically significant trends in generic component reliability data over periods of up to ten years. Therefore, no additional actions need to be performed to address this assessment for this application.

F&Os Regarding PRA Supporting Requirements

F&O ID:	Other Affected SRs:	Peer Review CC Assessment:
PRM-B11-01		Not Met

Capability Category I/II/III Requirements:

MODEL all operator actions and operator influences in accordance with the HRA element of this Standard.

F&O Issue and Proposed Resolution:

Median HEP values are utilized for the internal events related HFEs. Mean values are required.

Basis for Significance:

Potentially significant impact.

Possible Resolution:

Revise the HRA to model mean HEPs, rather than median HEPs. Consider documenting the internal events HRA in the HRA Calculator to be consistent with HEP development for the fire PRA.

Disposition of the Peer Review Finding:

Steps taken to resolve Peer Review assessment:

Duke has updated the HEP values as necessary to address the use of mean values. The updated HEP values have been incorporated into the Fire PRA quantification.

Evaluation of Peer Review assessment impact on proposed application:

The actions above resolve the issue and therefore, no additional actions are needed to address this F&O for this application.

F&Os Regarding PRA Supporting Requirements

F&O ID:	Other Affected SRs:	Peer Review CC Assessment:
FSS-C2-01		Met CC I

Capability Category II/III Requirements:

Characterize ignition source intensity using a realistic time-dependent fire growth profile for significant contributors as appropriate to the ignition source.

F&O Issue and Proposed Resolution:

Fire scenarios that expose external targets involving armored thermo set cabling with a thermoplastic coating are not assumed to spread/propagate beyond the initiator. A fire scenario CCDP is assigned to these scenarios based on loss of the initiator and the raceways located above the initiator within the ZOI. However the HRR for the scenario is not increased beyond that of the initiator. Based on discussions in NUREG/CR-6850, it is reasonable to conclude that fire will spread into these raceways and increase the HRR of the scenario due to burning cable. In scenarios where this is postulated by the generic fire modeling tools it would be reasonable to either assume room burnout (conservative approach), or perform detailed modeling given the increased HRR, to identify the fire damage state for the initiator/cable tray stack fire scenarios.

Fire scenario evaluation tools were developed based on generic fire modeling. These tools are based on a bounding fires that are assumed to burn at full peak intensity. The resultant evaluation tools are included in Attachment B of calculation MCC-1535.00-00-0104 (Draft). Because more realistic modeling was not used this SR is met at CC I.

Basis for Significance:

This finding is significant because the fire modeling to date may not have adequately characterized scenario HRR which impacts the identified fire damage state. The suggested increase in HRR has the potential to impact both plume and HGL scenarios.

Possible Resolution:

Expand fire scenarios that predict target fire damage to include HRR associated with cable fires. Alternatively severity factors may be applied to these scenarios such that low HRR fires are screened out and higher HRR are calculated as causing full room burnout

Disposition of the Peer Review Finding:

Steps taken to resolve Peer Review assessment:

This issue has been adequately addressed while responding to NRC Requests for Additional Information (RAIs) regarding the MNS License Amendment Request to implement a risk-informed performance-based fire protection program (i.e., NFPA-805). In Fire Modeling (FM) RAI 01.j, the NRC staff requested a re-evaluation of the effect of horizontal flame spread and vertical fire propagation in cable trays and the resulting increase HRR on the ZOI and Hot Gas Layer (HGL) analysis. In FM RAI 01.j.01, the NRC staff requested a re-quantification of the impact of fire propagation in cable trays and the HRR contribution from the cables on target damage and plant risk in light of the fact that the response to FM RAI 02.b indicated that 24% of the total cable population in the plant is thermoplastic.

In response to FM RAI 01.j.01 [Reference 40], which was answered in conjunction with the response to RAI 02.b.01 (requesting that the following sensitivity analysis take into account not only the thermoplastic damage criteria of target cables, but also model fire propagation in horizontal trays), a sensitivity analysis was performed comparing the HRR margin to a HRR that includes the contribution from all impacted cable trays. Where this margin was found to not be adequate, a full room burnout scenario was added to the PRA RAI 03 re-quantification. Additionally, a conservative evaluation of the impact of fire propagation in cable trays was performed. When added to the impact of thermoplastic cable insulation, the calculated increase in plant risk was approximately 1.74E-06/year CDF and 3.02E-07/year LERF for Unit 1, and approximately 1.39E-06/year CDF and 2.65E-07/year LERF for Unit 2.

Evaluation of Peer Review assessment impact on proposed application:

The analysis performed in response to the NFPA 805 RAI demonstrated the change in risk is small and the associated fire analysis changes performed in the RAI response have been incorporated into the Fire PRA analysis for the ILRT extension and as a result the impact of this F&O is directly calculated in the delta LERF analysis.

F&Os Regarding PRA Supporting Requirements

F&O ID:	Other Affected SRs:	Peer Review CC Assessment:
FSS-C5-01		Not Met

Capability Category I/II Requirements:

JUSTIFY that the damage criteria used in the Fire PRA are representative of the damage targets associated with each fire scenario.

F&O Issue and Proposed Resolution:

The MNS FPRA uses thermo set damage criteria for all cable damage criteria. The most common cable type in the plant is armored thermo set cable with a thermoplastic coating. Based on the discussion in NUREG/CR-6850, Section H.1.3 certain configurations of this type of cable exhibit thermoplastic damage criteria due to fire from the thermoplastic coating engulfing the armored cables. This damage occurs because the thermo set cable is exposed to flame temperatures which are well above the maximum thermo set damage temperatures (i.e., > 915 Deg F for > 1 min. Ref. NUREG/CR-6850 Table H-5). Discussion of this issue in calculation MCC-1535.00-00-0104 (Draft), Section 6.1 indicates that pooling of the thermoplastic material is not expected in any configuration however disposition of the potential for thermoplastic ignition and subsequent damage to thermoplastic insulated circuits should be addressed to fully justify this position.

Basis for Significance:

This finding could affect the generic Zone-Of-Influence used to perform screening of ignition sources. If thermoplastic damage criteria are required, the ZOI must be expanded to encompass this damage.

Possible Resolution:

Investigate piloted ignition temperature of thermoplastic materials and incorporate into justification for thermoset damage criteria.

Disposition of the Peer Review Finding:

Steps taken to resolve Peer Review assessment:

This issue has been adequately addressed while responding to NRC Requests for Additional Information (RAIs) regarding the MNS License Amendment Request to implement a risk-informed performance-based fire protection program (i.e., NFPA-805). In Fire Modeling (FM) RAI 01.j, the NRC staff requested a re-evaluation of the effect of horizontal flame spread and vertical fire propagation in cable trays and the resulting increase HRR on the ZOI and Hot Gas Layer (HGL) analysis. In FM RAI 02.b, the NRC staff requested that the cable targets that are characterized as thermoset (TS) are correctly assigned TS damage thresholds: i.e., that MNS confirm that the cables in question do have TS insulation.

In FM RAI 01.j.01, the NRC staff requested a re-quantification of the impact of fire propagation in cable trays and the HRR contribution from the cables on target damage and

plant risk in light of the fact that the response to FM RAI 02.b indicated that 24% of the total cable population in the plant is thermoplastic. In FM RAI 02.b.01, the NRC staff requested that the sensitivity analysis performed for FM RAI 01.j account for not only the thermoplastic damage criteria of target cables, but also model fire propagation in horizontal trays.

In response to these RAIs, FM RAI 01.j.01 and RAI 02.b.01 were answered in conjunction by performing a sensitivity analysis comparing the HRR margin to a HRR that includes the contribution from all impacted cable trays. Where this margin was found to not be adequate, a full room burnout scenario was added to the PRA RAI 03 re-quantification. Additionally, a conservative evaluation of the impact of fire propagation in cable trays was performed. When added to the impact of thermoplastic cable insulation, the calculated increase in plant risk was approximately 1.74E-06/year CDF and 3.02E-07/year LERF for Unit 1, and approximately 1.39E-06/year CDF and 2.65E-07/year LERF for Unit 2.

Evaluation of Peer Review assessment impact on proposed application:
The analysis performed in response to the NFPA 805 RAI demonstrated the change in risk is small and the associated fire analysis changes performed in the RAI response have been incorporated into the Fire PRA analysis for the ILRT extension and as a result the impact of this F&O is directly calculated in the delta LERF analysis.

F&Os Regarding PRA Supporting Requirements

F&O ID:	Other Affected SRs:	Peer Review CC Assessment:
FSS-C5-02		Not Met

Capability Category I/II Requirements:

JUSTIFY that the damage criteria used in the Fire PRA are representative of the damage targets associated with each fire scenario.

F&O Issue and Proposed Resolution:

The MNS FPRA uses thermoset damage criteria for all cable damage criteria. The most common cable type in the plant is armored thermoset cable with a thermoplastic coating. In addition to these cables, thermoplastic cables are included in some raceways and locations (Reference cables routed through Control Room floor). Raceways that include both thermoplastic and thermoset cables should have a thermoplastic damage criteria applied because failure and ignition of the thermoplastic cable will lead to rapid failure of the adjacent thermoset cables (Ref. NUREG/CR-6850, Section H.1.3).

Basis for Significance:

This finding could affect the generic Zone-Of-Influence used to perform screening of ignition sources. If thermoplastic damage criteria are required, the ZOI must be expanded to encompass this damage.

Possible Resolution:

The absence of thermoplastic cables in cable raceways that credit thermo set damage criteria should be verified to justify use of those criteria.

Disposition of the Peer Review Finding:

Steps taken to resolve Peer Review assessment:

This issue has been adequately addressed while responding to NRC Requests for Additional Information (RAIs) regarding the MNS License Amendment Request to implement a risk-informed performance-based fire protection program (i.e., NFPA-805). In Probability Risk Assessment (PRA) RAI 01.b, the NRC staff requested that the functions associated with non-armored cables used at MNS be identified, and for a basis for concluding that fire damage to these cables would not impact the Fire PRA results. In PRA RAI 03.b.01, the NRC staff requested a summary of the disposition of each issue in the aggregate PRA and the post-transition PRA. In response to these RAIs, the circuit failure probabilities for identified non-armored cables were updated to appropriate NUREG/CR-7150 values.

Additionally, as documented in the response to F&O CSS-C5-01, a conservative evaluation of the impact of fire propagation in cable trays was performed. When added to the impact of thermoplastic cable insulation, the calculated increase in plant risk was approximately 1.74E-06/year CDF and 3.02E-07/year LERF for Unit 1, and approximately 1.39E-06/year CDF and 2.65E-07/year LERF for Unit 2.

Evaluation of Peer Review assessment impact on proposed application:

The analysis performed in response to the NFPA 805 RAI demonstrated the change in risk is small and the associated fire analysis changes performed in the RAI response have been incorporated into the Fire PRA analysis for the ILRT extension and as a result the impact of this F&O is directly calculated in the delta LERF analysis.

F&Os Regarding PRA Supporting Requirements

F&O ID:	Other Affected SRs:	Peer Review CC Assessment:
FSS-D3-01		Met CC I

Capability Category II Requirements:

For any physical analysis unit that represents a significant contributor to fire risk, SELECT and APPLY fire modeling tools such that the scenario analysis provides reasonable assurance that the fire risk contribution of each unscreened physical analysis unit can be either bounded or realistically characterized.

F&O Issue and Proposed Resolution:

The zone of influence identified in the battery room includes some potentially non-conservative credit for the panel above the vital inverters (1EVID and 2EVID) as a radiant energy shield. Consideration of damage to adjacent panels (1EVID and 2EVID again) may be overly conservative given the location of the vents for these panels (at the top of the panels). Additional basis for credit for the heat shield and other deviations from the Generic Treatments specified in the Scenario Report, in the battery room and other areas, needs to be provided. The basis for not overlapping damage to abutting electrical cabinets (i.e., switchgear cubicles) to ensure all potential damage scenarios are enveloped requires further justification (e.g. FA 15-17, scenarios B4 & B6).

Basis for Significance:

The basis for deviations from the Generic Treatments and Scenario Report is not sufficiently supported in the documentation reviewed.

Possible Resolution:

Review and provide basis for any scenarios which may credit an approach deviating from the documented fire modeling bases (Generic Treatment). Additionally, potentially over-conservative damage to adjacent panels (1EVID and 2EVID again) was identified. Review of instances of this type should be conducted to prevent addition of unnecessary conservatism into the Fire PRA.

Disposition of the Peer Review Finding:

Steps taken to resolve Peer Review assessment:

The covers above the inverters and battery chargers are no longer credited as radiant energy shields preventing target damage in the MNS Fire PRA. It is common to create a single fire scenario to address both Unit 1 & 2 ignition sources simultaneously. During quantification of the Unit 1 Fire PRA, the Unit 2 failure(s) do not significantly contribute to the fire risk. During quantification of the Unit 2 Fire PRA, the Unit 1 failure(s) do not significantly contribute. This approach does not result in over-estimating damage.

Each set of switchgear cubicles that formed a unique scenario has a corresponding zone of influence with the potential to lead to overlapping target damage. Since most of the trays run parallel to the switchgear, the raceway nodes between scenario B4 and B5 and between B5 and B6 can be expected to be different. But the corresponding cable damage does overlap.

Evaluation of Peer Review assessment impact on proposed application:

Changes to the Fire PRA that are described in this F&O are sufficient to perform an analysis for the ILRT extension. Therefore, no additional actions are needed to address this F&O for this application.

F&Os Regarding PRA Supporting Requirements

F&O ID:	Other Affected SRs:	Peer Review CC Assessment:
HRA-A1-01		Not Met

Capability Category I/II/III Requirements:

For each fire scenario, for each safe shutdown action carried over from the Internal Events PRA, DETERMINE whether or not each action remains relevant and valid in the context of the Fire PRA consistent with the scope of selected equipment per the ES element and plant response model per the PRM element of this Standard, and in accordance with HLR-HR-E and its SRs in Part 2 with the
following clarifications:

(a) Where SR HR-E1 mentions "in the context of the accident scenarios," specific attention is to be given to the fact these are fire scenarios

and

(b) DEVELOP a defined basis to support the claim of nonapplicability of any of the requirements under HLR-HRE in Part 2.

F&O Issue and Proposed Resolution:

AP- 45 R7 Enclosure 1 (& Enclosure 2), Step 3 Instructs the operator to isolate all 3 PZR PORVs with the NOTE: Step 3 is time-critical and must be completed within 10 minutes of declaring the fire active. While this 10 min timeframe may be from the Hot Short Assumption, the timing on RNCBLKVDHE from the internal events PRA is 60 minutes based upon a single PORV failing open. Due to MSO, multiple PORVs could potentially fail open for these Fire Areas and there has been no accounting for this potential change in timing due to 2 or 3 PORVs failing open. Note: The resolution of this issue may impact the evaluation of success criteria, accident sequences, and the HRA SRs under HLR PRM-B.

Summary of Assessment:

At least one human action was found that the timing from the internal events model would not be applicable for all fire scenarios in an identified enclosure. No adjustment for this timing change was made in the HEP. Additionally, the lack of internal events pre-initiators and the use of median vs. mean values from the internal events PRA will requires some level of reanalysis and requantification of the fire CDF once incorporated. See F&Os PRM-B2-01 and PRM-B11-01

Basis for Significance:

New fire effect on HRA not considered. Additionally, the IE timing causes the HEP to screen to a lower multiplier even though the timing the action is applied to would be less than 60 minutes.

Possible Resolution:

Create HRA calculator analysis for fire induced combinations of PORV failure to reclose.

Disposition of the Peer Review Finding:

Steps taken to resolve Peer Review assessment:

The Fire PRA Model Report has been updated to document the review of the quantification of basic event RNCBLKVDHE which revealed that the HEP is not sensitive to the time available even with 3 PORVs open based on MAAP runs. Consequently, no adjustment to the HEP for this action is required for the possibility of multiple spurious PORV operation.

The Fire PRA incudes the same pre-initiator human events as the internal events PRA. Pre initiator human errors are included for the standby systems and trains. The most significant of these have been quantified based on the procedures while the less significant retain screening values. Duke has updated the HEP values as necessary to address the use of mean values. The updated HEP values have been incorporated into the Fire PRA quantification. This is more fully explained in the response to F&O PRM-B11-01.

Evaluation of Peer Review assessment impact on proposed application:

Mean HEPs have been used in the Fire HRA. These changes are included in the calculation analysis to determine the risk change in LERF for the ILRT extension. Therefore, no additional actions need be performed to address this assessment for this application.

F&Os Regarding PRA Supporting Requirements

F&O ID:	Other Affected SRs:	Peer Review CC Assessment:
HRA-A2-01		Met

Capability Category I/II/III Requirements:

For each fire scenario, IDENTIFY any new fire-specific safe shutdown actions called out in the plant fire response procedures (e.g., de-energizing equipment per a fire procedure for a specific fire location) in a manner consistent with the scope of selected equipment from the ES and PRM elements of the RA-S-2009 standard and in accordance with HLR-HR-E and its SRs in Part 2 with the following clarifications:

(a) Where SR HR-E1 discusses procedures, this is to be extended to procedures for responding to fires

(b) Where SR HR-E1 mentions "in the context of the accident scenarios," specific attention is to be given to the fact that these are fire scenarios

(c) Another source for SR-HR-E1 is likely to be the current Fire Safe Shutdown / Appendix R analysis

and

DEVELOP a defined basis to support the claim of nonapplicability of any of the requirements under HLR-HR-E in Part 2.

F&O Issue and Proposed Resolution:

The listing of the failures in the Summary document and the Appendix B listing of Operator actions did not match either each other or the CAFT A model or recovery file.

Basis for Significance:

Not having a consistent list of HEPs that are developed and used in the model raises questions about the completeness and accuracy of the evaluation. Are items missing that are believed to have been developed? Are items in the model that should have been removed from earlier evaluations (FIREFLDRHE)?

Possible Resolution:

Consolidate and verify multiple listings. Develop and document consistent fire HRA in Summary document, Operator Action Appendices, CAFT A model, and Recovery Files.

Disposition of the Peer Review Finding:

Steps taken to resolve Peer Review assessment:

Inconsistencies among the HFE evaluation document, Appendix B of the Component Selection calculation and the CAFTA model were reconciled.

Evaluation of Peer Review assessment impact on proposed application:

Changes to the HRA that are described in this F&O are sufficient to perform an analysis on the change in LERF for the ILRT extension. Therefore, no additional actions need be performed to address this assessment for this application.

F&Os Regarding PRA Supporting Requirements

F&O ID:	Other Affected SRs:	Peer Review CC Assessment:
HRA-A2-02		Met

Capability Category I/II/III Requirements:

For each fire scenario, IDENTIFY any new fire-specific safe shutdown actions called out in the plant fire response procedures (e.g., de-energizing equipment per a fire procedure for a specific fire location) in a manner consistent with the scope of selected equipment from the ES and PRM elements of the RA-S-2009 standard and in accordance with HLR-HR-E and its SRs in Part 2 with the following clarifications:

(a) Where SR HR-E1 discusses procedures, this is to be extended to procedures for responding to fires

(b) Where SR HR-E1 mentions "in the context of the accident scenarios," specific attention is to be given to the fact that these are fire scenarios

(c) Another source for SR-HR-E1 is likely to be the current Fire Safe Shutdown / Appendix R analysis

and

DEVELOP a defined basis to support the claim of nonapplicability of any of the requirements under HLR-HR-E in Part 2.

F&O Issue and Proposed Resolution:

New operator action developed in support of the fire PRA, FCACAST2DHE, instructs operators to close valves, (1CA7 AC and 1CA9B), from the MCR. In the fire response procedure AP-45 for Enclosures 5 (Step 4), Enclosure 7 (Step 3) et. al., these valves are opened and the breakers opened to prevent the valve changing state. This prior procedural action and the impact on the subsequent HEP does not appear to have been incorporated.

Basis for Significance:

The actions taken by the operators in AP-45 are time critical and should be evaluated for their impact on the associated fire HRAs.

Possible Resolution:

Update FCACAST2DHE to address plant equipment state at time action will need to be taken.

Disposition of the Peer Review Finding:

Steps taken to resolve Peer Review assessment:

The Model Development Report was revised to rename to FCACA02DHE and the HRA was addressed as an X-CR (outside control room) action, which also resolves the time action analysis.

Evaluation of Peer Review assessment impact on proposed application:

Changes to the HRA that are described in this F&O are sufficient to perform an analysis on the change in LERF for the ILRT extension. Therefore, no additional actions need be performed to address this assessment for this application

F&Os Regarding PRA Supporting Requirements

F&O ID:	Other Affected SRs:	Peer Review CC Assessment:
HRA-C1-01		Met CC I

Capability Category II Requirements:

For each selected fire scenario, QUANTIFY the HEPs for all HFEs, accident sequences that survive initial quantification and ACCOUNT FOR relevant fire-related effects using conservative estimate (e.g., screening values), in accordance with the SRs for HLR-HR-G in Part 2 set forth under Capability Category I, with the following clarifications:

(a) Attention is to be given to how the fire situation alters any previous assessments in nonfire analyses as to the influencing factors and the timing considerations covered in SRs HR-G3, HR-G4, and HR-G5 in Part 2

and

(b) DEVELOP a defined basis to support the claim of nonapplicability of any of the requirements under HLR-HR-G in Part 2.

F&O Issue and Proposed Resolution:

Methodology used for development of dependent human action failure probabilities is not standard and is not referenced. Standard methodology for determining failure rate for dependent human errors is described in detail in NUREG/CR-1278 Chapter 7.

Summary of Assessment:

Screening values used for multiple individual HFEs following quantification. All Joint Human Error Probabilities (JHEPs) are calculated using some portion of screening values. Dependency analysis does not use a standard methodology for calculating combined HEPs, all except the single lowest value HEP are replaced with screening values in calculation. The SR is judged to be met at CC I.

Basis for Significance:

Methodology used for development of HEPs replaces all of the HEPs in the model except for individual failures therefore could have significant impact of fire CDF.

Possible Resolution:

Use NUREG/CR-1278 methodology. EPRI HRA Calculator has automated this methodology for CAFTA cutsets; therefore, completion of Internal Events Analysis in HRA Calculator would further facilitate this process.

Disposition of the Peer Review Finding:

Steps taken to resolve Peer Review assessment:

The Finding was addressed by recalculating the joint dependencies for the internal events PRA using methodology consistent with NUREG/CR-1278. Important operator actions are reviewed to ensure the use of the HEP multipliers (fire adjustments) for individual HFEs do not introduce over-conservatism.

Additionally, one aspect of this issue has been adequately addressed while responding to NRC Requests for Additional Information (RAIs) regarding the MNS License Amendment Request to implement a risk-informed performance-based fire protection program (i.e., NFPA-805). In Probability Risk Assessment (PRA) RAI 01.c.iii, the NRC staff requested that each Joint HEP (JHEP) value used in the Fire PRA with a value below 1E-5 include its own justification that demonstrates the inapplicability of the NUREG-1792 guideline, which states that JHEP values should not be below 1E-5. Providing justification for the JHEPs less than 1E-5 in an RAI response addressed this particular concern.

The Fire HRA analysis has been revised to be consistent with NUREG-1921 approach and the multipliers have been removed.

Evaluation of Peer Review assessment impact on proposed application:

Changes to the HRA analysis that are described in this F&O resolution are incorporated directly into the ILRT extension determination of the increase in LERF risk. Therefore, no additional actions need be performed to address this assessment for this application.

F&Os Regarding PRA Supporting Requirements		
F&O ID:	**Other Affected SRs:**	**Peer Review CC Assessment:**
HRA-E1-01		Not Met
Capability Category I/II/III Requirements:		
DOCUMENT the Fire PRA HRA including (a) those fire-related influences that affect the methods, processes, or assumptions used as well as the identification and quantification of the HFEs/HEPs in accordance with HLR-HR-I and its SRs in Part 2, and develop a defined basis to support the claim of non-applicability of any of the requirements under HLR-HR-I in Part 2, *and* (b) any defined bases to support the claim of non-applicability of any of the referenced requirements in Part 2 beyond that already covered by the clarifications in the Part.		
F&O Issue and Proposed Resolution:		

Methodology for the HEP screening described in the documentation was not used in the development of the Appendix B Operator Action Review.

Summary of Assessment:

There is no traceable path from the documented definition of screening criteria to the documented HEP values for use in the FPRA.

Basis for Significance:

There is no current documentation of the Criteria used to develop increased HEPs in Appendix B for use in the fire PRA.

Possible Resolution:

Document a consistent process for use in screening and implement this consistently.

Disposition of the Peer Review Finding:

Steps taken to resolve Peer Review assessment:

This issue has been adequately addressed while responding to NRC Requests for Additional Information (RAIs) regarding the MNS License Amendment Request to implement a risk-informed performance-based fire protection program (i.e., NFPA-805). In Probability Risk Assessment (PRA) RAI 01.c, the NRC staff requested that the use of the multiplier methodology for developing human error probabilities (HEPs) and joint HEPs (JHEPs) be justified, or that the human reliability analysis (HRA) be revised to utilize HEP/JHEP values developed using NRC accepted methods such as NUREG-1921. Applying the NUREG-1921 methodology in the HRA addressed this particular concern.

In the final aggregate analysis of the Fire PRA, the HEP multipliers were replaced with a combination of fire-specific quantified HEP values and NUREG-1921 screening criteria values. A subsequent revision to the Fire PRA Model Development Calculation (MCC-1535.00-00-105) incorporated the changes made in response to PRA RAI 01.c.

Evaluation of Peer Review assessment impact on proposed application:

Changes to the HRA analysis that are described in this F&O resolution are incorporated directly into the ILRT extension determination of the increase in LERF risk. Therefore, no additional actions need be performed to address this assessment for this application.

F&Os Regarding PRA Supporting Requirements

F&O ID:	Other Affected SRs:	Peer Review CC Assessment:
SF-A2-01		Not Met

Capability Category I/II/III Requirements:

For those physical analysis units within the Fire PRA global plant analysis boundary,

(a) REVIEW installed fire detection and suppression systems and provide a qualitative assessment of the potential for either failure (e.g., rupture or unavailability) or spurious operation during an earthquake,

and

(b) ASSESS the potential impact of system rupture or spurious operation on post-earthquake plant response including the potential for flooding relative to water-based fire suppression system, loss of habitability for gaseous suppression systems, and the potential for diversion of suppressants from areas where they might be needed for those fire suppression systems associated with a common suppressant supply

F&O Issue and Proposed Resolution:

Requirement addressed by walkdown review with the exception of loss of habitability and suppression system diversion.

Basis for Significance:

Qualitative evaluation specified and not performed or documented.

Possible Resolution:

Perform and/or document evaluation.

Disposition of the Peer Review Finding:

Steps taken to resolve Peer Review assessment:

No impact on quantification (seismic-fire interaction is purely qualitative per NUREG/CR-6850). The Fire PRA Summary Report has been updated to address habitability impacts beyond what was captured in the IPEEE documentation. The Halon cylinders are located in each Turbine Building basement. In the unlikely event of an earthquake causing a cylinder to rupture, the Halon would be dissipated over the very large volume of the Turbine Building and the resulting concentration levels would not be expected to significantly impact habitability.

Evaluation of Peer Review assessment impact on proposed application:

The review and assessment performed for this F&O resolution is not impacted by, and sufficiently addresses, the ILRT extension. Therefore, no additional actions need to be performed to address this assessment for this application.

F&Os Regarding PRA Supporting Requirements

F&O ID:	Other Affected SRs:	Peer Review CC Assessment:
SF-A3-01		Not Met

Capability Category I/II/III Requirements:

ASSESS the potential for common-cause failure of multiple fire suppression systems due to the seismically induced failure of supporting systems such as fire pumps, fire water storage tanks, yard mains, gaseous suppression storage tanks, or building standpipes.

F&O Issue and Proposed Resolution:

This SR is not directly addressed in the Fire PRA.

MCS-1465 00-00-0008, Rev. 9 (FP DBD) indicates (Section C.18.4) that fire pumps are located in a non-seismic Cat III intake. Not clear if this indicates a potential common mode failure.

Basis for Significance:

Indication in documentation of potential common mode seismic impact to Fire Pumps.

A reference to procedures for recovering from a loss of fire water system pumps was provided (OP/0/B/6400/002 D) and appears to address the specific concern identified above. An evaluation of other potential common cause losses of fire system equipment appears to be necessary to meet this requirement.

Possible Resolution:

Evaluate potential seismic common mode failures of fire protection systems.

Disposition of the Peer Review Finding:

Steps taken to resolve Peer Review assessment:

No impact on quantification (seismic-fire interaction is purely qualitative per NUREG/CR-6850). The Fire Protection Specification notes that the fire pumps are located in a non-seismic structure. In the event that the fire pumps are disabled, TSC Volume 2, Enclosures 46 & 47 provide for the deployment of a portable (Hale) pump or use of the CAST to pressurize the RY header if necessary for fire suppression (stated in the Summary Report, section 3.13).

Evaluation of Peer Review assessment impact on proposed application:

The assessment performed for this F&O resolution is not impacted by, and sufficiently addresses, the ILRT extension. Therefore, no additional actions need to be performed to address this assessment for this application.

F&Os Regarding PRA Supporting Requirements		
F&O ID:	**Other Affected SRs:**	**Peer Review CC Assessment:**
SF-A4-01		Not Met
Capability Category I/II/III Requirements:		
REVIEW plant seismic response procedures *and* Qualitatively ASSESS the potential that a seismically induced fire, or the spurious operation of fire suppression systems, might compromise post-earthquake plant response.		
F&O Issue and Proposed Resolution:		
Could not locate evaluation of potential fire in plant seismic response procedure. **Basis for Significance:** Procedural link could not be found. **Possible Resolution:** Add review for fire in procedure.		
Disposition of the Peer Review Finding:		
Steps taken to resolve Peer Review assessment: No impact on quantification (seismic-fire interaction is purely qualitative per NUREG/CR-6850). It is noted that Procedure RP/0/A/5700/007, "Earthquake" does not reference fire response procedure AP-45 or TSC Volume 2, Enclosure 46 & 47; however, in the event of a seismic induced fire it is expected that multiple procedures will be used in parallel as necessary. The entry conditions for the fire response procedure (via fire alarm annunciator or report of a fire) apply at all times and under any plant operating conditions. **Evaluation of Peer Review assessment impact on proposed application:** The assessment performed for this F&O resolution is not impacted by, and sufficiently addresses, the ILRT extension. Therefore, no additional actions need to be performed to address this assessment for this application.		

F&Os Regarding PRA Supporting Requirements

F&O ID:	Other Affected SRs:	Peer Review CC Assessment:
SF-A5-01		Not Met

Capability Category I/II/III Requirements:

REVIEW

(a) Plant fire brigade training procedures and ASSESS the extent to which training has prepared firefighting personnel to respond to potential fire alarms and fires in the wake of an earthquake

and

(b) The storage and placement of firefighting support equipment and fire brigade access routes,

and

(c) ASSESS the potential that an earthquake might compromise one or more of these features

F&O Issue and Proposed Resolution:

Could not locate fire brigade training requirements with respect to seismic / fire interaction.

Basis for Significance:

Brigade training requirement could not be located.

Possible Resolution:

Address seismic / fire interaction in brigade training.

Disposition of the Peer Review Finding:

Steps taken to resolve Peer Review assessment:

No impact on quantification of Fire PRA or Change Evaluations: seismic-fire interaction is purely qualitative per NUREG/CR-6850.

Evaluation of Peer Review assessment impact on proposed application:

The assessment performed for this F&O resolution is not impacted by, and sufficiently addresses, the ILRT extension. Therefore, no additional actions need to be performed to address this assessment for this application.

F&Os Regarding PRA Supporting Requirements

F&O ID:	Other Affected SRs:	Peer Review CC Assessment:
MU-B3-01		Met

Capability Category I/II/III Requirements:

Changes to a PRA due to PRA maintenance and PRA upgrade shall meet the requirements of the Technical Requirements Section of each respective Part of this Standard.

F&O Issue and Proposed Resolution:

In their PRA configuration control procedure, Duke indicates the PRA updates / upgrades must be performed in accordance with Section 4 of the ASME PRA Standard, RA-Sb-2005. The referenced standard only covered at-power internal initiators. This standard has been replaced by ASME/ ANS RA-Sa-2009 which covers at-power internal initiators, internal fires and external events. It is this standard which is endorsed by the NRC in the latest revision of RG 1.200

Basis for Significance:

Duke is referencing an out-of -date standard the only covers internal events.

Possible Resolution:

Update the reference in XSAA-I 06 to point to ASME/ANS RA-Sa-2009. Portions of XSAA-106 pertaining to internal initiators and internal flood should reference Sections 2 and 3 respectively and the portions of XSAA-106 pertaining to internal fires should reference section 4.

Disposition of the Peer Review Finding:

Steps taken to resolve Peer Review assessment:

This reference has been updated by Duke Energy, in procedure AD-NF-NGO-0500, Corporate PRA Support For Emergent Issues And Risk-Informed Applications.

Evaluation of Peer Review assessment impact on proposed application:

The assessment performed for this F&O resolution is not impacted by, and sufficiently addresses, the ILRT extension. Therefore, no additional actions need to be performed to address this assessment for this application.

F&Os Regarding PRA Supporting Requirements		
SR ID:	**Other Affected SRs:**	**Peer Review CC Assessment:**
PP-B3 (note, no F&O)		Met CC I

Capability Category II/III Requirements:

If spatial separation is credited as a partitioning feature, JUSTIFY the judgment that spatial separation is sufficient to substantially contain the damaging effects of any fire that might be postulated in each of the fire compartments created as a result of crediting this feature.

Issue and Proposed Resolution:

Volume 2, Chapter 1 of NUREG/CR-6850, TR-1011989 [4-1] discusses criteria that may be applied in justifying decisions related to spatial separation as a partitioning feature. Section 7.1 of plant partitioning and ignition frequency calculation states "The fire compartments were mapped directly to fire areas; therefore no crediting of partitioning features that do meet the formal NUREG/CR-6850 criteria for compartments was applied. Consequently, MNS 'compartments' are enclosed rooms with rated fire barriers."

Possible Resolution:

Update Fire PRA analysis to meet Capability Category II/III Requirements.

Disposition of the Peer Review Finding:

Steps taken to resolve Peer Review assessment:

None. Compartmentalization based on Fire Area boundaries is adequate for plant partitioning without the need to credit spatial separation. Therefore, spatial separation is not credited as a partitioning feature in the Fire PRA, and no further action is required. CC I for this SR is acceptable for the Fire PRA analysis.

Evaluation of Peer Review assessment impact on proposed application:

CC I for this Fire PRA SR is sufficient for this ILRT extension application, given that spatial separation is not credited in the analysis.

F&Os Regarding PRA Supporting Requirements		
SR ID:	**Other Affected SRs:**	**Peer Review CC Assessment:**
PP-B5 (note, no F&O)		Met CC I

Capability Category II/III Requirements:

DEFINE and JUSTIFY the basis and criteria applied when active fire barrier elements (such as normally open fire doors, water curtains, and fire dampers) are credited in partitioning.

Issue and Proposed Resolution:

Volume 2, Chapter 1 of NUREG/CR-6850, TR-1011989 [4-1] discusses criteria that may be applied in justifying decisions related to active fire barrier elements as a partitioning feature. Section 7.1 of plant partitioning and ignition frequency calculation states "The fire compartments were mapped directly to fire areas; therefore no crediting of partitioning features that do meet the formal NUREG/CR-6850 criteria for compartments was applied. Consequently, MNS 'compartments' are enclosed rooms with rated fire barriers."

Possible Resolution:

Update Fire PRA analysis to meet Capability Category II/III Requirements.

Disposition of the Peer Review Finding:

Steps taken to resolve Peer Review assessment:

None. Compartmentalization based on Fire Area boundaries is adequate for plant partitioning without the need to credit active fire barrier elements. Therefore, active fire barrier elements are not credited as a partitioning feature in the Fire PRA, and no further action is required. CC I for this SR is acceptable for the Fire PRA analysis.

Evaluation of Peer Review assessment impact on proposed application:

CC I for this Fire PRA SR is sufficient for this ILRT extension application given that no active fire barriers are credited in the analysis.

F&Os Regarding PRA Supporting Requirements		
SR ID:	**Other Affected SRs:**	**Peer Review CC Assessment:**
FSS-C1 (note, no F&O)		Met CC I

Capability Category II Requirements:

For each selected fire scenario, ASSIGN characteristics to the ignition source using a two-point fire intensity model that encompass low likelihood, but potentially risk contribution, fire events in the context of both fire intensity and duration given the nature of the fire ignition sources present.

Issue and Proposed Resolution:

Fire modeling was conducted via generic fire modeling from which Zones-of-Influence (ZOI) for specific initiator types was generated. The ZOIs were used to define bounding fire characteristics for each fire scenario. Fire scenarios were not considered to spread beyond the initial ZOI due to the typical construction of cable in the plant (i.e., armored thermo set cable with a thermoplastic coating).

Because bounding fire initiator characteristics are assigned for each scenario this SR is met at CC I.

Possible Resolution:

Update Fire PRA analysis to meet Capability Category II Requirements.

Disposition of the Peer Review Finding:

Steps taken to resolve Peer Review assessment:

The Fire PRA analysis was updated to increase the number of scenario refinements using a 2-point treatment, consistent with the requirement of CC II.

Evaluation of Peer Review assessment impact on proposed application:

The use of two-point fire model has been incorporated into the analysis calculating the change in fire risk LERF for the ILRT extension.

F&Os Regarding PRA Supporting Requirements

SR ID:	Other Affected SRs:	Peer Review CC Assessment:
FSS-C4 (note, no F&O)		Met CC I

Capability Category II Requirements:

If a severity factor is credited in the analysis, ENSURE that

(a) the severity factor remains independent of other quantification factors
(b) the severity factor reflects the fire event set used to estimate fire frequency
(c) the severity factor reflects the conditions and assumptions of the specific fire scenarios under analysis, and
(d) a technical basis supporting the severity factor's determination is provided

Issue and Proposed Resolution:

Calculation MCC-1535.00-00-0104 (Draft), identifies severity factors for a number of fire initiators in the analysis. These severity factors are based on the results of generic fire modeling; accordingly they provide bounding for the conditions and assumptions of specific scenarios.

Possible Resolution:
Update Fire PRA analysis to meet Capability Category II Requirements.

Disposition of the Peer Review Finding:

Steps taken to resolve Peer Review assessment:

Calculation MCC-1535.00-00-0104 identifies severity factors for a number of fire initiators in the analysis. These severity factors are based on NUREG/CR-6850 HRR distribution profiles. Severity 'alignment' factors based on observed experience which provided a practical alternative to the conservative HRR values for the less severe fires have been removed.

The phrase "conditions and assumptions of the specific fire scenarios under analysis" refers to those characteristics of the fire scenario that could influence whether or not a fire will damage targets. The intent of FSS-C4 for Capability Categories II and III is, in part, that such factors would be an explicit consideration in quantifying the severity factor. The intent for Capability Category I is to allow for the application of generic severity factors that reflect, more generally, those fire events that contributed to the fire ignition frequency but without explicit consideration of such case-specific factors so long as the severity factor applied is consistent with, and independent of, other quantification.

Evaluation of Peer Review assessment impact on proposed application:

CC I for this Fire PRA SR is sufficient for this ILRT extension application. The use of conservatively bounding severity factors based on generic fire modeling, instead of severity factors that reflect the conditions of the specific scenarios under analysis, will lead to a slightly conservative result.

F&Os Regarding PRA Supporting Requirements		
SR ID:	**Other Affected SRs:**	**Peer Review CC Assessment:**
FSS-D7 (note, no F&O)		Met CC I

Capability Category II Requirements:

In crediting fire detection and suppression systems, USE generic estimates of total system unavailability provided that

(a) the credited system is installed and maintained in accordance with applicable codes and standards

(b) the credited system is in a fully operable state during plant operation,

and

(c) The system has not experienced outlier behavior relative to system unavailability

Issue and Proposed Resolution:

MNS FPRA uses NUREG/CR-6850 detection system reliability.

Possible Resolution:

Update Fire PRA analysis to meet Capability Category II Requirements.

Disposition of the Peer Review Finding:

Steps taken to resolve Peer Review assessment:

Fire system availability/unavailability is not specifically addressed. Limited credit for suppression was taken in the analysis. Availability / unavailability is not applicable to the credit for prompt suppression for the hotwork fire scenarios: hotwork activities are accompanied by a fire watch and rely on manual suppression. Credit for manual or automatic suppression was taken in the multi-compartment analysis (MCA); however, the MCA is a screening analysis and is largely unaffected by the additional precision associated with suppression system unavailability.

Typical Fire PRA practice involves the application of a non-suppression probability, that is, the probability that suppression efforts fail to suppress the fire prior to the onset of the postulated equipment/cable damage. Hence, the non-suppression probability estimate includes an assessment of effectiveness (including the relative timing of fire damage versus detection/suppression and fire brigade performance), discussed in FSS-D8, as well as an overall assessment of system unavailability. The intent of SR FSS-D7 is to require increasing levels of plant specificity in assessing system unavailability with increasing capability category.

This issue has been further addressed while responding to NRC Requests for Additional Information (RAIs) regarding the MNS License Amendment Request to implement a risk-informed performance-based fire protection program (i.e., NFPA-805). In Probability Risk Assessment (PRA) RAI 01.d, the NRC staff requested that justification be provided for using

the generic unavailability estimates for the Halon systems credited in the evaluation of the turbine-driven auxiliary feedwater pump fire scenarios, which are taken from NUREG/CR-6850, bound actual system unavailability. The response to this RAI states that automatic suppression credit is taken for the Halon systems in the Unit 1 and Unit 2 Turbine Driven Auxiliary Feedwater Pump rooms by applying a NUREG/CR-6850 generic unreliability factor of 0.05. A review of the MNS Fire Impairment Log records (from February, 2012 to September, 2014) confirmed that the unavailability of the Unit 1 and Unit 2 Auxiliary Feedwater Pump turbine Halon 1301 fire suppression systems was less than 0.05, validating the appropriateness of using the NUREG/CR-6850 generic unreliability factor.

Evaluation of Peer Review assessment impact on proposed application:

Analyses done to validate the Fire PRA, which are described in this assessment, are sufficient to perform an analysis on the ILRT extension. Therefore, no additional actions need to be performed to address this assessment for this application.

F&Os Regarding PRA Supporting Requirements

SR ID:	Other Affected SRs:	Peer Review CC Assessment:
FSS-E3 (note, no F&O)		Met CC I

Capability Category II Requirements:

PROVIDE a mean value of, and statistical representation of, the uncertainty intervals for the parameters used for modeling the significant fire scenarios.

F&O Issue and Proposed Resolution:

Generic fire modeling treatments address uncertainties associated with the analysis.

Possible Resolution:

Update Fire PRA analysis to meet Capability Category II Requirements.

Disposition of the Peer Review Finding:

Steps taken to resolve Peer Review assessment:

N/A. CC I was considered acceptable for this SR, as methods for developing the statistical representation of the uncertainty intervals and mean values did not exist at the time of the Fire PRA peer review. In general, Generic Fire Modeling Treatments bound uncertainties associated with the Fire PRA analysis.

This issue has been further addressed while responding to NRC Requests for Additional Information (RAIs) regarding the MNS License Amendment Request to implement a risk-informed performance-based fire protection program (i.e., NFPA 805). In Fire Modeling (FM) RAI 02.a, the NRC staff requested a description of how the uncertainty associated with the fire model input parameters was accounted for in the analyses. The response to this RAI provided an explanation that parameter uncertainties are addressed through the use of conservative and bounding analyses and by performing sensitivity studies. Detailed

discussions were provided to explain how this was accomplished for the three primary FM activities for which parameter uncertainty is applicable: MCR abandonment analysis, HGL tabulations and ZOI tabulations. Parametric uncertainty was adequately accounted for in the FM analysis performed in support of the NFPA 805 transition.

Evaluation of Peer Review assessment impact on proposed application:

CC I was considered acceptable for this SR, as methods for developing the statistical representation of the uncertainty intervals and mean values did not exist at the time of the Fire PRA peer review. In general, Generic Fire Modeling Treatments bound uncertainties associated with the Fire PRA analysis. This issue has been further addressed while responding to NRC Requests for Additional Information (RAIs) regarding the MNS License Amendment Request to implement a risk-informed performance-based fire protection program (i.e., NFPA 805). CC I for this Fire PRA SR is sufficient for this ILRT extension application.

F&Os Regarding PRA Supporting Requirements

SR ID:	Other Affected SRs:	Peer Review CC Assessment:
FSS-F3 (note, no F&O)		Met CC I

Capability Category II/III Requirements:

If, per SR FSS-F1, one or more scenarios are selected, COMPLETE a quantitative assessment of the risk of the selected fire scenarios in a manner consistent with the FQ requirements, including collapse of the exposed structural steel.

F&O Issue and Proposed Resolution:

No quantitative evaluation performed (Fire Scenario Report Section 3.2).

Possible Resolution:

Update Fire PRA analysis to meet Capability Category II Requirements.

Disposition of the Peer Review Finding:

Steps taken to resolve Peer Review assessment:

N/A. CC I was considered acceptable for this SR. The Fire PRA locations were reviewed and determined to not meet the definition in FSS-F1. Therefore, this SR is N/A.

Evaluation of Peer Review assessment impact on proposed application:

The Fire PRA locations were reviewed and determined to not meet the definition in FSS-F1. Therefore, this SR is N/A, and this Fire PRA SR is sufficient for this ILRT extension application.

F&Os Regarding PRA Supporting Requirements		
SR ID:	**Other Affected SRs:**	**Peer Review CC Assessment:**
FSS-H2 (note, no F&O)		Met CC I
Capability Category II/III Requirements:		
DOCUMENT a basis for target damage mechanisms and thresholds used in the analysis, including references for any plant-specific or target-specific performance criteria applied in the analysis.		
F&O Issue and Proposed Resolution:		
The Hughes report "Generic Fire Modeling Treatments," prepared for ERIN, Project Number 1SPH02902.030, is the basis for thresholds used to identify the fire targets. While the application is plant-specific, the report itself is generic. **Possible Resolution:** Update Fire PRA analysis to meet Capability Category II Requirements.		
Disposition of the Peer Review Finding:		
Steps taken to resolve Peer Review assessment: N/A. CC I was considered acceptable for this SR. The damage thresholds applied in the Generic Fire Modeling Treatments are taken from NUREG/CR-6850. No plant specific data is available for use in lieu of NUREG/CR-6850. Since the plant specific ignition sources are comparable to those in the Generic Fire Modeling Treatments, use of zone of influence information based on the generic configurations is considered acceptable. **Evaluation of Peer Review assessment impact on proposed application:** CC I for this Fire PRA SR is sufficient for this ILRT extension application. Since the plant-specific ignition sources are comparable to those in the Generic Fire Modeling Treatments, use of zone of influence information based on the generic configurations is considered acceptable and conservative.		

F&Os Regarding PRA Supporting Requirements		
F&O ID:	**Other Affected SRs:**	**Peer Review CC Assessment:**
HRA-B4 (note, no finding-level F&O)		Met CC I (note, there are no requirements for CC I)
Capability Category II Requirements:		

INCLUDE HFEs for cases where fire-induced instrumentation failure of any single instrument could cause an undesired operator action, consistent with HLR-ES-C of this Part and in accordance with HLR-HR-F and its SRs in Part 2

and

DEVELOP a defined basis to support the claim of nonapplicability of any of the requirements under HLR-HR-F in Part 2.

F&O Issue and Proposed Resolution:

No detailed HEPs developed for instrumentation failures.

Possible Resolution:

HEPs for instrumentation failures would need to be developed.

Disposition of the Peer Review Finding:

Steps taken to resolve Peer Review assessment:

Subsequent to the Fire PRA peer review, the Fire PRA Component Selection calculation was completed and approved. As stated in this calculation, "[n]o single instruments were identified that would cause an undesired operator action without first taking one or more confirmatory actions. All Operations staff respond to routine plant duties and to plant events based on confirmatory indications such as redundant instrumentation and independent measurements..."

Since no cases were identified where fire-induced instrumentation failure of any single instrument could cause an undesired operator action, no additional or specific HFE analysis is required to meet the CC II requirements of this SR. Therefore, this SR is considered to be met at CC II.

This issue has been further addressed while responding to NRC Requests for Additional Information (RAIs) regarding the MNS License Amendment Request to implement a risk-informed performance-based fire protection program (i.e., NFPA 805). In PRA RAI 01.e, the NRC staff requested that the basis for concluding that SR HRA-B4 "..is now considered met at CC II" be clarified. The HRA included a review of potential errors of commission due to fire-induced faulty instrumentation readings, and determined that there were no instances where an undesired operator action would be taken without first taking one or more confirmatory actions.

Evaluation of Peer Review assessment impact on proposed application:

The stated disposition for this Fire PRA SR is sufficient for this ILRT extension application. No aspect of the ILRT extension analysis is impacted by any fire-induced instrumentation failure that would affect operator actions that would be addressed by this SR.