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DTE Energy

10 CFR 50.90

March 22, 2016
NRC-16-0006

U. S. Nuclear Regulatory Commission
Attention: Document Control Desk
Washington, D.C. 20555-0001

References:
1) Fermi 2
NRC Docket No. 50-341
NRC License No. NPF-43

2) Nuclear Energy Institute (NEI) Topical Report, 94-01, Revision 3-A, "Industry Guideline for Implementing Performance-Based Option of 10 CFR Part 50, Appendix J," July 2012 (ADAMS Accession No. ML12221A202)

3) NRC Regulatory Guide 1.163, Revision 0, "Performance Based Containment Leak-Test Program," September 1995 (ADAMS Accession No. ML003740058)

4) Nuclear Energy Institute (NEI) Topical Report, 94-01, Revision 2-A, "Industry Guideline for Implementing Performance-Based Option of 10 CFR Part 50, Appendix J," October 2008 (ADAMS Accession No. ML100620847)

5) NRC Letter, "Fermi-2 – Issuance of Amendment Re: Implementation of 10 CFR Part 50 Appendix J Option B," August 8, 1996 (ADAMS Accession No. ML020730597)

Subject: License Amendment Request to Revise Integrated Leak Rate Test (Type A) and Type C Test Intervals

In accordance with the provisions of Title 10 *Code of Federal Regulations* (10 CFR) 50.90, DTE Electric Company (DTE) requests an amendment to the Fermi 2 Plant Operating License, Appendix A, Technical Specifications (TS), to allow for permanent extension of the Type A primary containment integrated leak rate test (ILRT) interval to 15 years and extension of the Type C test interval up to 75 months, based on acceptable

performance history as defined in approved Nuclear Energy Institute (NEI) Topical Report 94-01, Revision 3-A "Industry Guideline for Implementing Performance-Based Option of 10 CFR Part 50, Appendix J," (Reference 2).

The requested amendment would revise TS Section 5.5.12, "Primary Containment Leakage Rate Testing Program", by replacing the reference to Regulatory Guide (RG) 1.163 "Performance-Based Containment Leak-Test Program," dated September 1995 (Reference 3), with a reference to follow guidance in NEI 94-01, Rev 3-A, and conditions and limitations for ILRT testing from NRC's Safety Evaluation Report (SER) for NEI 94-01, Revision 2-A (Reference 4). NEI 94-01 Revisions 2-A and 3-A were approved by the NRC to describe an acceptable approach for implementing the optional performance-based requirements of Option B to 10 CFR 50, Appendix J, as modified by the conditions and limitations in the respective SERs.

Additionally, the proposed change would allow the extension of the containment isolation valve leakage test (Type C) from its current 60 month frequency to 75 months in accordance with NEI 94-01, Rev 3-A. The proposed change would also adopt a more conservative grace interval of 9 months as opposed to the current 15 month interval for Type B and Type C tests in accordance with NEI 94-01, Rev 3-A. The technical analysis for the proposed amendment is consistent with the guidance in Section 9.2.3 of NEI 94-01, Rev 3-A.

This license amendment request also proposes two administrative changes. The first change revises TS 5.5.12 to remove a one-time extension of the Type A test frequency from 2007. This extension is complete and is no longer needed to be reflected in TS.

The second administrative change will revise the Fermi 2 Operating License, Section D, to remove a reference to an exemption regarding Appendix J testing of containment air locks. TS Amendment 108, approved by the NRC in 1996 (Reference 5), eliminated the need for this exemption by revising the testing program for containment air locks.

Enclosure 1 provides a description and assessment of the proposed change. Enclosure 2 provides the existing Operating License and TS pages marked up to show the proposed change. Enclosure 3 provides revised (clean) Operating License and TS pages. There are no TS Bases associated with TS 5.5.12. Enclosure 4 provides the probabilistic risk assessment evaluation for the proposed change.

DTE requests approval of the proposed License Amendment by January 31, 2017 to support preparations for the 18th Refueling Outage, currently scheduled to start on March 20, 2017. The amendment will be implemented within 60 days.

No new commitments are being made in this submittal.

In accordance with 10 CFR 50.91, a copy of this application, with enclosures, is being provided to the designated Michigan State Official.

Should you have any questions or require additional information, please contact Mr. Alan I. Hassoun, Manager, Licensing at (734) 586-4287.

I declare under penalty of perjury that the foregoing is true and correct.

Executed on March 22, 2016

Keith J. Polson
Site Vice President

Enclosures:
1. Evaluation of the Proposed License Amendment
2. Marked-up Pages of Existing Fermi 2 Operating License and TS
3. Clean Pages of Fermi 2 Operating License and TS with Changes Incorporated
4. Probabilistic Risk Assessment Report

cc: Director, Office of Nuclear Reactor Regulation
NRC Project Manager
NRC Resident Office
Reactor Projects Chief, Branch 5, Region III
Regional Administrator, Region III
Michigan Public Service Commission
Regulated Energy Division (kindschl@michigan.gov)

Enclosure 1 to
NRC-16-0006

Fermi 2 NRC Docket No. 50-341
Operating License No. NPF-43

Evaluation of the Proposed License Amendment

Evaluation of the Proposed License Amendment

Contents

Acronym List

10 CFR	Title 10 *Code of Federal Regulations*
AF	As-found
AL	As-left
ANS	American Nuclear Society
ANSI	American National Standards Institute
ASME	American Society of Mechanical Engineers
BWR	Boiling Water Reactor
CCFP	Conditional Containment Failure Probability
CDF	Core Damage Frequency
CRD	Control Rod Drive
DER	Deviation Event Report
DTE	DTE Electric Company
ECCS	Emergency Core Cooling System
EPRI	Electric Power Research Institute
GE	General Electric
HPCI	High Pressure Core Injection
ILRT	Integrated Leak Rate Test
IN	Information Notice
IPE	Individual Plant Examination
IPEEE	Individual Plant Examination of External Events
ISI	Inservice Inspection
IWE	Inservice Inspection Program
LER	Licensee Event Report
LERF	Large Early Release Frequency
LLRT	Local Leakage Rate Test
LOCA	Loss of Coolant Accident
MSIV	Main Steam Isolation Valve
NDE	Nondestructive Examination
NEI	Nuclear Energy Institute
NRC	Nuclear Regulatory Commission
PRA	Probabilistic Risk Assessment
RCIC	Reactor Core Isolation Cooling
RF##	Refueling Outage (##)
RG	Regulatory Guide
RHR	Residual Heat Removal
RPV	Reactor Pressure Vessel
SCFH	Standard Cubic Feet per Hour
SER	Safety Evaluation Report
SRV	Safety Relief Valve
TIP	Traverse In-core Probe
TR	Topical Report
TS	Technical Specifications
UCL	Upper Confidence Limit
UT	Ultrasonic Testing

1.0 SUMMARY DESCRIPTION

In accordance with the provisions of Title 10 *Code of Federal Regulations* (10 CFR) 50.90, DTE Electric Company (DTE), requests an amendment to the Fermi 2 Plant Operating License, Appendix A, Technical Specifications (TS), to allow for permanent extension of the Type A primary containment integrated leak rate test (ILRT) test interval to 15 years and permanent extension of the Type C test interval up to 75 months, based on acceptable performance history as defined in Nuclear Energy Institute (NEI) Topical Report (TR) 94-01, Revision 3-A, "Industry Guideline for Implementing Performance-Based Option of 10 CFR Part 50, Appendix J" (Reference 6.1).

The requested amendment would revise TS Section 5.5.12, "Primary Containment Leakage Rate Testing Program," by replacing the reference to Regulatory Guide (RG) 1.163 "Performance-Based Containment Leak-Test Program," dated September 1995 (Reference 6.2), with a reference to follow guidance in NEI 94-01, Revision 3-A, and conditions and limitations specified in NEI 94-01, Revision 2-A (Reference 6.3), that was found acceptable by the Nuclear Regulatory Commission (NRC) to describe an acceptable approach for implementing the optional performance-based requirements of Option B to 10 CFR 50, Appendix J, as modified by the conditions and limitations in the NRC's Safety Evaluation Report (SER) (included in Reference 6.3). This license amendment request also proposes administrative changes to TS 5.5.12 and Operating License Provision D.

The purpose of NEI 94-01 guidance is to assist licensees in the implementation of Option B to 10 CFR 50, Appendix J, "Leakage Rate Testing of Containment of Light Water Cooled Nuclear Power Plants," (hereafter referred to as Appendix J, Option B). Revision 2-A of NEI 94-01 added guidance for extending containment ILRT Type A test surveillance intervals beyond ten years; and Revision 3-A of NEI 94-01 added guidance for extending containment isolation valve (Type C test) local leakage rate test (LLRT) surveillance intervals beyond 60 months.

The technical basis for the proposed license amendment utilizes risk-informed analysis augmented with non-risk related considerations. A risk impact evaluation, provided in Enclosure 4, concluded that the increases in large early release frequency (LERF) are within the limits set forth by the applicable guidance contained in RG 1.174 (Reference 6.4), NUREG-1493 "Performance-Based Containment Leak-Test Program" (Reference 6.5), and Electric Power Research Institute (EPRI) Report No. 1009325 Rev-2A (Reference 6.6).

In accordance with the guidance of NEI 94-01 Revision 3-A, DTE proposes to extend the maximum surveillance interval for the ILRT to no longer than 15 years from the last ILRT based on satisfactory performance history. The current interval is no longer than 10 years and would require that the next ILRT for Fermi 2 be performed during the Spring 2017 refueling outage (RF18). The proposed change would allow the ILRT Type A to be performed on or before November 7, 2022. DTE has applied for a renewal of the Fermi 2 Operating License (Reference 6.7). This amendment will reduce the number of Type A tests performed over the period of extended operation resulting in significant savings in radiation exposure to personnel.

The proposed change would also allow the extension of the containment isolation valve leakage test (Type C) from its current 60 month frequency to 75 months in accordance with NEI 94-01, Revision 3-A. The proposed change would also adopt a more conservative grace interval of 9 months as opposed to the current 15 month interval for Type B and Type C tests in accordance with NEI 94-01, Revision 3-A. The technical analysis for the proposed amendment is consistent with the guidance in Section 9.2.3 of NEI 94-01, Revision 3-A. The proposed change will also delete a historical reference in TS 5.5.12.a that refers to the first Type A test after October 1992 being performed no later than October 2007.

Additionally, the proposed amendment will remove the exemption for testing containment air locks in the Fermi 2 Operating License, Provision D (labelled "c" in the list of exemptions under provision D). The need for this exemption was eliminated when Amendment 108 was approved on August 8, 1996 (Reference 6.8); however, Amendment 108 did not remove this text from the Operating License.

2.0 Detailed Description

2.1 Description of Proposed Change

The proposed license amendment would revise TS Section 5.5.12.a by changing the wording to indicate that the program shall be in accordance with NEI 94-01, Revision 3-A, instead of NRC Regulatory Guide 1.163.

Current TS Section 5.5.12.a states in part that:

> *This program shall be in accordance with the guidelines contained in Regulatory Guide 1.163, "Performance-Based Containment Leak-Test Program, "dated September 1995, with the exception of approved exemptions to 10 CFR 50, Appendix J, and as modified by the following exception to NEI 94-01, Rev. 0, "Industry Guideline for Implementing Performance-Based Option of 10 CFR 50, Appendix J":*
>
> *The first Type A test after the October 1992 test shall be performed no later than October 2007.*

The proposed amendment would remove the outdated reference to the 1992 and 2007 Type A tests and revise the TS to state:

> *This program shall be in accordance with Nuclear Energy Institute (NEI) 94-01, Revision 3-A, "Industry Guideline for Implementing Performance-Based Option of 10 CFR Part 50, Appendix J," dated July 2012, and the limitations and conditions specified in NEI 94-01, Revision 2-A, dated October 2008.*

Provision D of the Fermi 2 Operating License states in part:

> *Exemptions from certain requirements of Appendices E and J to 10 CFR Part 50, are described in supplements to the SER. These include: (a) an exemption from the requirement of Section IV.F of Appendix E that a full participation emergency planning exercise be conducted within one year before issuance of the first operating license for full power and prior to operation above five percent of rated power (Section 13.3 of SSER #6); (b) an exemption from the requirement of Paragraph III.C.2(b) of Appendix J, the testing of the main steam isolation valves at the peak calculated containment pressure associated with the design basis accident (Section 6.2.7 of SSER #5); and (c) an exemption from the requirement of Paragraph III.D.2(b)(ii) of Appendix J, the testing of containment air locks at times when containment integrity is not required (Section 6.2.7 of SSER #5).*

This amendment would remove the exemption for air lock testing and revise this section to state:

> *Exemptions from certain requirements of Appendices E and J to 10 CFR Part 50, are described in supplements to the SER. These include: (a) an exemption from the requirement of Section IV.F of Appendix E that a full participation emergency planning exercise be conducted within one year before issuance of the first operating license for full power and prior to operation above five percent of rated power (Section 13.3 of SSER #6); and (b) an exemption from the requirement of Paragraph III.C.2(b) of Appendix J, the testing of the main steam isolation valves at the peak calculated containment pressure associated with the design basis accident (Section 6.2.7 of SSER #5).*

The need for the Operating License exemption for containment air locks was eliminated when the NRC approved TS Amendment 108 in 1996; however, this statement was not removed from the Operating License at that time.

A marked-up copy of the proposed changes is provided in Enclosure 2. Enclosure 3 provides revised (clean) pages. Enclosure 4 provides a plant specific risk analysis to revise the ILRT test interval.

2.2 Description of Fermi 2 Primary Containment

Fermi 2 utilizes a General Electric (GE) Boiling Water Reactor (BWR) Mark I primary containment structure. The containment consists of an inverted light-bulb-shaped steel liner vessel (drywell) and a torus-shaped suppression chamber. The primary containment consists of a drywell that houses the reactor pressure vessel (RPV), vent system connecting the drywell and the suppression chamber, reactor coolant recirculation system, vacuum relief system, isolation valves, and other primary components.

The drywell is a steel pressure vessel that is enclosed in reinforced concrete for shielding purposes. The drywell is separated from the reinforced concrete structure by a gap of

approximately 2 inches. This gap is filled with a compressible polyurethane material to allow for movement between the drywell and concrete. The polyurethane sheets are coated on both sides with an epoxy resin binder to prevent water leakage into the material. The bottom portion of the drywell shell is totally embedded in concrete. There is compacted sand at the lower portion of the transition zone between the upper freestanding section and the bottom embedded section of the drywell to allow for thermal expansion and to aid in the drainage of condensate that may accumulate in the 2-inch gap outside the Drywell. Four drain lines are used to remove any moisture in the sand cushion.

The drywell vessel is provided with a removable head to facilitate refueling. Two bolted equipment hatches are provided for access into the drywell and one control rod drive (CRD) removal hatch is used for CRD replacement. There is also one double-door personnel airlock. The locking mechanism on each airlock door is designed to maintain a tight seal when the doors are subject to either external or internal pressure. The doors are mechanically interlocked so that neither door can be operated unless the other door is closed and locked. The drywell head and equipment hatches are designed with double gaskets with intermediate leak taps provided for leak testing. Provisions have been made to leak test the personnel air lock and door seals, equipment hatches, and CRD hatch.

The Pressure Suppression Chamber, or torus, is a steel pressure vessel, in the shape of a torus, situated below and encircling the Drywell. The suppression chamber shell thickness is typically 0.587 inches above the horizontal centerline and 0.658 inches below the horizontal centerline, except at penetration locations where it is locally thicker. The Suppression Chamber is supported vertically by inside and outside columns and by a saddle support that spans the inside and outside columns. The support system transmits dead weight and seismic and hydrodynamic loading to the reinforced-concrete foundation slab of the reactor building. Space is provided outside the chamber for inspection. The Suppression Chamber has been subsequently reevaluated for the effects of LOCA-related loads and SRV discharge-related loads defined by NRC Safety Evaluation Report NUREG-0661 “Safety Evaluation Report – Mark I Containment Long-Term Program, Resolution of Generic Technical Activity A-7,” (Reference 6.9), and GE Reports NEDO-21888, ”Mark I Containment Program Load Definition Report,” (Reference 6.10) and NEDC-31897P-1, “Generic Guidelines for General Electric Boiling Water Reactor Power Uprate,” (Reference 6.11). There are two 4-feet diameter manhole entrances with double-gasket bolted covers connected to the chamber by 4-feet diameter steel pipes. They are designed with leak testing connections. These access ports are closed when Primary Containment is required and opened when pressure suppression capability is no longer required.

Eight circular vent pipes, 6 feet in diameter, form the connection between the drywell and the suppression chamber. These vent pipes are connected to a torus-shaped ring header, 4 feet, 3 inches in diameter, placed within the air space of the suppression chamber. The vent pipes are provided with two-ply expansion bellows to accommodate differential motion between the drywell and suppression chamber. These bellows have test connections that allow for leak testing and for determining that the passages between the two-ply bellows are not obstructed. Projecting downward from the torus-shaped ring header are 80 downcomer pipes, 24 inches in

diameter, projecting from the ring header and terminating below the water surface in the suppression chamber.

Primary containment penetrations carry piping, mechanical systems, and electrical wiring through the biological shield and primary containment vessel. These penetrations are classified as piping penetrations (sleeved and unsleeved), electrical service penetrations, mechanical system penetrations (traversing in-core probe penetrations), and access openings. The containment penetrations have the following design characteristics to maintain design containment integrity: Capability to withstand peak transient pressures, capability to withstand without failure the forces caused by impingement of the fluid from the rupture of the largest local pipe or connection, and the capability to accommodate without failure the thermal and mechanical stresses that may be encountered during all modes of operation.

2.3 Testing Requirement of 10 CFR 50, Appendix J

Testing requirements of 10 CFR 50, Appendix J, provide assurance that leakage from the containment, including systems and components that penetrate the containment, does not exceed the allowable leakage rates (values) specified in the TS. Periodic surveillances of isolation valves and reactor containment penetrations are performed so that proper repairs and maintenance are made during the service life of the systems and components penetrating primary containment and the containment itself. Appendix J identifies three types of tests that are required: (1) Type A tests, which are intended to measure the primary containment overall integrated leakage rate; (2) Type B tests, which are intended to detect local leaks and measure leakage across pressure-retaining leakage-limiting boundaries such as penetrations; (3) Type C tests, which are intended to measure containment isolation valve leakage rates. The Type A ensures containment structure integrity by evaluating those structural parts of containment that are not covered by Type B and Type C testing.

In 1995, 10 CFR 50 Appendix J was amended to provide a performance-based Option B for the containment leakage testing requirements. Option B requires that test intervals for Type A, B, and C testing be determined using a performance-based approach. Performance-based intervals are based on the performance history of the component and resulting risk from its failure. RG 1.163 was also issued in 1995. RG 1.163 endorsed NEI 94-01, Revision 0, "Industry Guideline for Implementing Performance-Based Option of 10 CFR Part 50, Appendix J," with certain modifications and additions. RG 1.163 and NEI 94-01, Revision 0, allow licensees with a satisfactory ILRT performance history to reduce the test frequency for the ILRT test from three tests in 10 years to one test in 10 years. This relaxation in testing frequency was based on an NRC risk assessment contained in NUREG-1493 "Performance-Based Containment Leak-Test Program," and EPRI TR-104285, "Risk Impact Assessment of Revised Containment Leak Rate Testing Intervals," (Reference 6.12), both of which showed that the risk increase associated with extending the ILRT surveillance interval was very small.

NEI submitted TR 94-01, Revision 2, and EPRI Report No. 1009325, Revision 2, "Risk Impact Assessment of Extended Integrated Leak Rate Testing Intervals," to the NRC for review. NEI 94-01, Revision 2, describes an acceptable approach for implementing the optional performance-

based requirements of Option B described in 10 CFR Part 50, Appendix J, which includes provisions for extending Type A ILRT intervals to 15 years and incorporates the regulatory positions in RG 1.163. It delineates a performance-based approach for determining Type A, B, and C containment leakage rate surveillance testing frequencies. This method uses industry performance data, plant-specific performance data, and risk insights in determining the appropriate testing frequency. It also discusses the performance factors that licensees must consider in determining test intervals. However, it does not address how to perform the tests because these details can be found in existing documents (e.g., ANSI/ANS-56.8-2002 (Reference 6.13)).

EPRI TR-1009325, Revision 2 (Reference 6.6), provides a risk impact assessment for optimized ILRT intervals of up to 15 years, utilizing current industry performance data and risk-informed guidance, primarily Revision 1 of RG 1.174, “An Approach for using Probabilistic Risk Assessment in Risk-Informed Decisions on Plant-Specific Changes to the Licensing Basis.” The NRC final SER (included in Reference 6.3), documents the NRC’s evaluation and acceptance of NEI 94-01, Revision 2, subject to the specific limitations and conditions listed in Section 4.1 of the SER, and the evaluation and acceptance of EPRI TR-1009325, Revision 2. The accepted version of NEI 94-01 has been issued as Revision 2-A, dated October 2008.

In 2012, NEI 94-01, Revision 3-A, was issued. This document describes an acceptable approach for implementing the optional performance-based requirements of Option B to 10 CFR Part 50, Appendix J, and includes provisions from NEI 94-01, Revision 2-A, for extending Type A ILRT intervals to 15 years. NEI 94-01, Revision 3-A, was approved by NRC (SER included in Reference 6.1) as an acceptable methodology for complying with the provisions of Option B to 10 CFR Part 50. It delineates a performance-based approach for determining Type A, B, and C containment leakage rate surveillance testing frequencies. Justification of extending test intervals is based on the performance history and risk insights. Extensions of Type B and Type C test intervals are allowed based upon completion of two consecutive periodic as-found tests where the results of each test are within a licensee's allowable administrative limits. Intervals may be increased from 30 months up to a maximum of 120 months for Type B tests (except for containment airlocks) and up to a maximum of 75 months for Type C tests. If a licensee considers extended test intervals of greater than 60 months for Type B or Type C tested components, the review should include the additional considerations of as-found tests; schedule; review of performance history, data analysis, risk-impact; and Type B and C leakage as described in NEI 94-01, Revision 3-A, Section 11.3.2.

The NRC clarified the use of grace in the deferral of ILRTs past the 15 year interval in NEI 94-01, Revision 2-A, NRC SER Section 3.1.1.2:

> "As noted above, Section 9.2.3, NEI TR 94-01, Revision 2, states, "Type A testing shall be performed during a period of reactor shutdown at a frequency of at least once per 15 years based on acceptable performance history." However, Section 9.1 states that the "required surveillance intervals for recommended Type A testing given in this section may be extended by up to 9 months to accommodate unforeseen emergent conditions but should not be used for routine scheduling and planning purposes." The NRC staff

> believes that extensions of the performance-based Type A test interval beyond the required 15 years should be infrequent and used only for compelling reasons.
>
> Therefore, if a licensee wants to use the provisions of Section 9.1 in TR NEI 94-01, Revision 2, the licensee will have to demonstrate to the NRC staff that an unforeseen emergent condition exists."

NEI 94-01, Revision 3-A, Section 10.1 concerning the use of grace in the deferral of Type B and Type C LLRTs past intervals of up to 120 months for the recommended surveillance frequency for Type B testing and up to 75 months for Type C testing, states that:

> "Consistent with standard scheduling practices for Technical Specifications Required Surveillances, intervals of up to 120 months for the recommended surveillance frequency for Type B testing and up to 75 months for Type C testing given in this section may be extended by up to 25% of the test interval, not to exceed nine months.
>
> Notes: For routine scheduling of tests at intervals over 60 months, refer to the additional requirements of Section 11.3.2, of NEI 94-01, Revision 3-A.
>
> Extensions of up to nine months (total maximum interval of 84 months for Type C tests) are permissible only for non-routine emergent conditions. This provision (nine month extension) does not apply to valves that are restricted and/or limited to 30 month intervals in Section 10.2 (such as BWR MSIVs) or to valves held to the base interval (30 months) due to unsatisfactory LLRT performance."

The NRC also provided the basis for extending ILRT intervals to 15 years in NEI 94-01, Revision 3-A, NRC SER Section 4.0:

> "The basis for acceptability of extending the ILRT interval out to once per 15 years was the enhanced and robust primary containment inspection program and the local leakage rate testing of penetrations. Most of the primary containment leakage experienced has been attributed to penetration leakage and penetrations are thought to be the most likely location of most containment leakage at any time."

2.4 <u>Fermi 2 Previous Submittals</u>

DTE requested changes to the Operating License and TS by application dated December 21, 1995. The request was to permit implementation of 10 CFR Part 50, Appendix J, Option B. This request was approved by the NRC as license Amendment 108 on August 8, 1996 (Reference 6.8).

DTE requested a TS change by application, dated May 23, 2002, to allow a one-time deferral of the Type A primary containment ILRT. TS 5.5.12, “Primary Containment Leakage Rate Testing Program,” was revised to extend the current interval for performing the containment Type A test to 15 years. This one-time extension was approved by the NRC as license Amendment 153 on

March 27, 2003 (Reference 6.14). The reference to this extension is being removed as part of this amendment request.

3.0 Technical Evaluation

3.1 Adoption of NEI 94-01, Revision 3-A and Revision 2-A

As required by 10 CFR 50.54(o), the Fermi 2 primary containment shall be subject to the requirements set forth in 10 CFR Part 50, Appendix J. Option B of Appendix J requires that test intervals for Type A, Type B and Type C testing be determined by using a performance-based approach. Currently, the Fermi 2 Appendix J Testing Program Plan is based on RG 1.163 (Reference 6.2), which endorses NEI 94-01, Revision 0, with certain modifications and additions. This license amendment request proposes to revise the Appendix J Testing Program Plan by implementing the guidance in NEI 94-01, Revision 3-A (Reference 6.1), and the conditions and limitations identified in NEI 94-01, Revision 2-A (Reference 6.3). Revision 2-A of NEI 94-01 describes an acceptable approach for implementing the optional performance-based requirements of Option B, including provisions for extending Type A (ILRT) intervals to 15 years. Revision 3-A of NEI 94-01 describes an acceptable approach for implementing the optional performance-based requirements of Option B, including provisions for extending Type B tests from 30 to 120 months, and extending Type C tests up to a maximum of 75 months. Both revisions incorporate the regulatory positions stated in RG 1.163 as modified by the NRC SERs of June 25, 2008 and June 8, 2012 for Revisions 2-A and 3-A of NEI 94-01, respectively (included in References 6.3 and 6.1, respectively).

NRC staff reviewed NEI 94-01, Revision 3-A, and determined that it describes an acceptable approach for implementing the optional performance-based requirements of Option B to 10 CFR Part 50, Appendix J, as modified by the conditions and limitations summarized in Section 4.0 of the SER. Table 1 addresses the two limitations and conditions for NEI 94-01, Revision 3-A, listed in Section 4.0 of the NRC safety evaluation.

In the SER issued by the NRC letter, dated June 25, 2008, the NRC concluded that NEI 94-01, Revision 2-A, describes an acceptable approach for implementing the optional performance-based requirements of Option B of 10 CFR Part 50, Appendix J, and found that NEI 94-01, Revision 2-A, is acceptable for referencing by licensees proposing to amend their TS in regards to containment leakage rate testing, subject to the limitations and conditions noted in Section 4.1 of the SER. Table 2 below addresses each of the six limitations and conditions for NEI 94-01, Revision 2-A, listed in Section 4.1 of the NRC SER.

Table 1: Limitations/Conditions from NEI 94-01, Revision 3-A

Limitation/Condition (from Section 4.0 of NRC SER, not quoted due to length)	DTE Response
Topical Report Condition 1	Following NRC approval of this license amendment request, DTE will include margin between the Type B and Type C leakage rate summation and its regulatory limit in the post-outage report and develop and maintain corrective actions to restore margin to acceptable level, as necessary. In addition, DTE will only extend the Type C test interval to 84 months for non-routine emergent conditions, subject to the limitations in Topical Report Condition 1.
Topical Report Condition 2	Following NRC approval of this license amendment request, if routinely scheduling the LLRT valve interval beyond 60-months and up to 75-months, DTE will include in the post-outage report an estimate of the amount of understatement in the Type B and C total, and the reasoning and determination of the acceptability of the extension to demonstrate that the LLRT totals calculated represent the actual leakage potential of the penetrations.

Table 2: Limitations/Conditions from NEI 94-01, Revision 2-A

Limitation/Condition (from Section 4.1 of NRC Safety Evaluation)	DTE Response
1. For calculating the Type A leakage rate, the licensee should use the definition in the NEI TR 94-01, Revision 2, in lieu of that in ANSI/ANS-56.8-2002.	Following NRC approval of this license amendment request, DTE will use the definition in NEI 94-01, Revision 3-A, for calculating the Type A leakage rate, which is the same definition as in NEI 94-01, Revision 2.
2. The licensee submits a schedule of containment inspections to be performed prior to and between Type A tests.	The frequency for performing general visual examinations of accessible interior and exterior surfaces of the containment for structural deterioration that may affect leak-tight integrity are performed as shown below in Table 5: ASME Class MC Examination Schedule for Fermi 2 (ASME Section XI, 2001 Edition with 2003 Addenda). The Fermi 2 IWE program is in its second 10 year interval.
3. The licensee addresses the areas of the containment structure potentially subjected to degradation.	General visual examinations (general visual, VT-3, and VT-1 (if required)) of accessible interior and exterior surfaces of the class MC components, parts and appurtenances of the containment as well as a visual examination of the moisture barrier at the concrete-to-steel interface are performed. These include inspections of the downcomers, ring girders, torus shell (interior/exterior), vent lines, penetration sleeves, bolting, coatings, and penetration bellows (which are leak rate tested). Fermi 2 has a sand cushion region. In 2013, all four sand cushion drain lines were internally inspected with a boroscope to ensure free water flow. No water was present and a drain path through the lines was verified. Periodic inspections of the drain lines are performed. There are currently no primary containment surface areas that require augmented examinations in accordance with ASME Section XI, Subsection IWE-1240.

Limitation/Condition (from Section 4.1 of NRC Safety Evaluation)	DTE Response
4. The licensee addresses any tests and inspections performed following major modifications to the containment structure, as applicable.	There are currently no planned or anticipated major modifications to the Fermi 2 containment structure. The station design change process would address testing requirements for any future containment structure modifications.
5. The normal Type A test should be less than 15 years. If a licensee has to utilize the provision of Section 9.1 of NEI TR 94-01, Revision 2, related to extending the ILRT interval beyond 15 years, the licensee must demonstrate to the NRC staff that it is an unforeseen emergent condition.	Fermi 2 acknowledges and accepts this NRC staff position, as communicated.
6. For plants licensed under 10 CFR Part 52, applications requesting a permanent extension of the ILRT surveillance interval to 15 years should be deferred until after the construction and testing of containments for that design have been completed and applicants have confirmed the applicability of NEI TR 94-01, Revision 2 and EPRI Report No. 1009325, Revision 2, including the use of past containment ILRT data.	Not applicable. Fermi 2 is not licensed under 10 CFR Part 52.

3.2 Leak Test History

3.2.1 Type A Testing

The historical results of the Type A tests for Fermi 2 are included in Table 3. The reported leakage rate is at the 95 percent upper confidence level and includes any Type B and Type C penalties (penetrations that are not vented).

The last Type A test was completed on November 7, 2007. Previous Type A testing confirmed that the containment structure leakage is acceptable, with considerable margin with respect to the TS acceptance criterion of 0.75 L_a, (where L_a is the maximum allowable leakage at pressure P_a (design basis loss of coolant accident pressure)). The calculated leakage rates at the 95 percent confidence level were below the acceptance criterion of 0.375 weight percent/day. The Appendix J acceptance criterion at 95 percent confidence level is 0.75 L_a = (0.75)(0.50 weight percent/day) = 0.375 weight percent/day at P_a. Since the last three Type A test results, as shown in Table 3, were less than 0.375 weight percent/day (0.75 L_a), a test frequency of at least once per 10 years was justified in accordance with NEI 94-01, Revision 0 (Reference 6.15).

Repair or replacement activities (including any unplanned activities) performed on the pressure retaining boundary of the primary containment prior to next scheduled Type A test would be

subject to the leakage test requirements of ASME Boiler and Pressure Vessel Code Section XI, Subsection IWE required examinations. There are no anticipated repairs or modifications of the containment that could affect leak tightness that would not be measured by local leakage rate testing as required in Section 9.2.4 of NEI 94-01, Revision 3-A.

Following the approval of this license amendment, the next Type A test for the Fermi 2 primary containment must be performed on or before November 7, 2022.

Table 3: DTE Electric Company Type A Test Historical Results Since 1989

Test Completion Date	Upper Confidence Limit Measured Leakage (%/day)	Correction for Type B and C Tests (%/day)	Total Leakage (%/day)	Acceptance Criteria (%/day)
11/29/1989	0.285	0.033	0.318	0.375
11/01/1992	0.212	0.032	0.244	0.375
11/10/2007	0.1168	0.0964	0.2132	0.375

The following is a description of the results from the latest two ILRTs at Fermi 2:

The November 2007 periodic Type A test was performed using BN-TOP-1 calculated at 95 percent upper confidence limit (UCL). The performance leak rate corresponding to the definition in NEI 94-01, Revision 0, was equal to the as-left results of 0.213 %/day.

The October 1992 periodic Type A test was performed using BN-TOP-1 calculated at 95 percent UCL, which resulted in a value of 0.212 %/day. The performance leak rate corresponding to the definition in NEI 94-01, Revision 0, was equal to the as-left results of 0.244 %/day.

As required by NEI 94-01, Revision 3-A, Section 9.1.2 "Test Interval," extensions on test intervals are allowed based on the requirements in Section 9.2.3 and two consecutive, periodic successful Type A tests. The results in Table 3 above show that there has been substantial margin to the maximum allowable leakage rate of 0.375 weight percent/day (0.75 L_a).

3.2.2 Type B and C Testing

The Type B and Type C containment leakage rate testing program for Fermi 2 requires tests intended to detect or measure leakage across pressure-retaining or leakage-limiting boundaries and containment isolation valves. As discussed in NUREG-1493 and NEI 94-01, Revision 3-A, Type B and Type C tests can identify the majority of all containment leakage paths. This amendment request adopts the guidance in NEI 94-01, Revision 3-A, in place of NEI 94-0, Revision 0.

As shown in Table 4, a review of the Type B and Type C test results from Spring 2003 through Fall of 2015 has generally shown a large amount of margin between the actual as-found (AF) and

as-left (AL) outage summations and the TS leakage rate acceptance criteria (that is, less than 0.6 L_a). Two Licensee Event Reports (LERs) were issued to document issues in 2006 and 2007.

Table 4: Fermi 2 Type B and Type C Leak Rate Summation History Since 2003

RF	RF09 (2003)	RF10 (2004)	RF11 (2006)	RF12 (2007)	RF13 (2009)	RF14 (2010)	RF15 (2012)	RF16 (2014)	RF17 (2015)
AF Min Path (scfh)	62.50	84.80	379.79 (Note 1)	359.83 (Note 2)	39.79	46.71	43.51	44.54	78.17
Percentage of 0.6 L_a	35.2%	47.7%	213.6% (Note 1)	202.4% (Note 2)	22.4%	26.3%	24.5%	25.1%	44.0%
AL Max Path (scfh)	74.83	62.32	67.20	51.49	75.48	76.94	80.38	88.91	74.46
Percentage of 0.6 L_a	42.1%	35.1%	37.8%	29.0%	42.5%	43.3%	45.2%	50.0%	41.9%

scfh = standard cubic feet per hour
Note 1: LER 2006-001, "Excessive Feedwater Check Valve Leakage at Containment Penetration" was issued (Reference 6.16).
Note 2: LER 2007-001, "Excessive Feedwater Check Valve Leakage at Containment Penetration" was issued (Reference 6.17).

LER 2006-001 was issued to document the penetration (X-9A) minimum-pathway air leakage value was greater than the allowable containment leakage rate (La) value of 296.3 scfh per TS. The failure was attributed to soft seat degradation on the feedwater line check valves. For both the inboard and outboard check valves, the soft seats were replaced and the soft seat service time was limited to two operating cycles. For the inboard check valve, the internal shaft was replaced and the alignment between the disc and the valve seat was adjusted.

LER 2007-001 was issued to document a containment minimum-pathway leak rate for a reactor feedwater line (Penetration X-9B) that exceeded the limiting conditions for operation in the plant TS. The valve failures were primarily attributed to soft seat erosion by feedwater flow to the point that the seats were not providing an effective seal. The corrective actions described in LER 2006-001 were taken for penetration X-9A, and this failure occurred before additional actions for other penetrations were implemented. A preventive maintenance frequency to replace the soft seats of one refuel outage was established for the outboard feedwater check valves. In addition, the inboard valves were hard seated in RF14 (October 2010 - December 2010), and have been successfully as-found tested to present.

Tables 4.1-1 and 4.1-2 identify the components that have not demonstrated acceptable performance during the previous two outages for Fermi 2.

Type B tests include: electrical penetrations, vacuum breaker-electrical penetrations, drywell head, equipment hatches, personnel airlock (interior seal, exterior seal, equalizing valve), reactor vessel stabilization manhole O-ring seals, expansion bellows, TIP flanges, RHR test line orifices, RHR spool flange, butterfly valve flange, torus water management supply spool flanges, two-ply torus bellows, relief valve flanges, and RHR blind flanges. The percent of the total number of

Type B tested components that are on a 120-month extended performance-based test interval is 93.6% (102 of 109). Seven penetrations cannot be placed on the Option B program (Personnel Airlock, Drywell Head, two Equipment Hatches, CRD Hatch, and two Torus Hatches).

Type C tests include primary containment isolation valves. The percent of the total number of Type C tested components that are on a 60-month extended performance-based test interval is 75.9% (151 of 199). Twenty-eight valves cannot be placed in the Option B program (MSIVs, Purge and Vent valves, and Feedwater Check valves).

Table 4.1-1: Type B and Type C LLRT Program Implementation Review of RF16 (2014)

Penetration No.	Component	As-Found (SCFH)	Admin Limit (SCFH)	As-Left (SCFH)	Cause of Failure	Corrective Action	Scheduled Interval
X-42	C4100F007	Leakage Off Scale	3.00	0.17	Valve seating surface included seat scratches, wear, particular, build up.	Maintenance performed. Replaced disc (soft seat).	30 month
X-19	G1154F018	5.75	2.00	5.75	Poor water quality, which has led to seat degradation.	Evaluation, required testing in RF17.	30 month
X-19	G1100F019	2.30	2.00	2.30	Poor water quality, which has led to seat degradation.	Evaluation, required testing in RF17.	30 month
X-10	E5150F008	As-Found test not performed	2.00	0.05	Steam cut on the bonnet sealing surface.	Maintenance performed. Replaced valve bonnet.	30 month
X-9B	B2100F076B	11.66	10.0	0.05	Valve seat condition.	Maintenance performed. Replaced soft seat.	30 month

Table 4.1-2: Type B and Type C LLRT Program Implementation Review of RF17 (2015)

Penetration No.	Component	As-Found (SCFH)	Admin Limit (SCFH)	As-Left (SCFH)	Cause of Failure	Corrective Action	Scheduled Interval
X-42	C4100F006	Leakage Off Scale	3.00	0.91	Valve seating surface included seat scratches, wear, particular, build up.	Maintenance performed. Disassembled valve and cleaned, measured, inspected and replaced soft seat.	30 month
X-42	C4100F007	16.30	3.00	0.10	Seat degradation	Maintenance performed. Replaced seat.	30 month
X-19	G1100F019	7.32	2.00	0.05	Not identified	Maintenance performed. Wedge polished.	30 month
X-18	G1154F600	10.69	4.50	0.41	Degraded gasket and packing.	Maintenance performed. Replaced gasket, packing and gland follower.	30 month
X-205B	T2300F450A	32.03	20.00	2.86	Valve seat/seal condition (missing washers)	Maintenance performed. Cleaned pallet seat, replaced seal and missing washers.	30 month
X-18	G1100F003	15.55	2.00	0.06	Metal shavings/rust identified during valve inspection	Maintenance performed. Cleaned valve internals. New packing.	30 month
X-25	T4803F602	Leakage Off Scale	14.87	0.95	Valve seat condition	Maintenance performed. Cleaned seat.	184 days

3.3 Containment Inspections

The ILRT, LLRTs, and Inservice Inspection (ISI) program for the primary containment collectively ensure the leak-tightness and structural integrity of the containment. The Containment Inservice Inspection Program implements the requirements of 10 CFR 50.55a. Fermi 2 follows the ASME Section XI, Subsection IWE, 2001 Edition with the 2003 Addenda, as mandated and modified by 10 CFR 50.55a for the current 10 year interval. Visual examinations are required per TS 5.5.6 and TS surveillance SR 3.6.1.1.1, and leakage rate testing is required per TS 5.5.12. Fermi 2 falls under ASME Section XI, Subsection IWE and not Subsection IWL.

General visual examinations of the accessible surfaces of primary containment are performed in accordance with the Primary Containment Inservice Inspection Program. These examinations are performed to assess the general condition of the primary containment surfaces and to satisfy the visual examination requirements of ASME Section XI, Subsection IWE. These examinations are performed in sufficient detail to detect signs of degradation. Detailed visual examinations are performed to determine the magnitude and extent of degradation of suspect surfaces initially detected by general visual examinations. The conditions reported during the examinations are evaluated to determine acceptability. The conditions are acceptable if it is determined that there is no evidence of damage or degradation sufficient to warrant further evaluation or performance of repair and replacement activities.

3.3.1 Containment Inservice Inspection Program (IWE)

The scope of the Fermi 2 IWE program includes the free-standing steel containment vessel and its integral attachments, containment hatches, airlocks, moisture barrier, and pressure-retaining bolting. The components subject to examination are described in the Fermi 2 ISI-NDE Program. These include accessible surface areas, bolted connections, accessible surface areas of wetted or submerged, the vent system accessible areas, moisture barriers, and coatings. In Subsection IWE, each 10 year ISI interval is divided into three approximately equal-duration inspection periods, and the examinations are conducted during the refueling outages within these inspections periods. A general visual inspection of 100 percent of accessible surface areas of the metallic containment is performed each inspection period. In addition, a general visual inspection of 100 percent of the moisture barrier is performed each inspection period. Since a 15 year ILRT interval spans at least four IWE inspection periods, the performance of examinations in accordance with Subsection IWE assures that at least three general visual examinations of containment pressure boundary, and the moisture barrier will be conducted during refueling outages between Type A tests if the Type A test interval is extended to 15 years. The frequency of examinations performed in accordance with Subsection IWE satisfies the requirement of NEI 94-01, Revision 3-A, Section 9.2.3.2.

At Fermi 2, Carboline Carbo Zinc 11 is utilized for the interior and exterior of the drywell, and the interior surfaces of the Suppression Chamber including the vent system are coated with Plastite 7155. The full length of the exterior surface of the drywell is coated with Carboline Carbo Zinc 11. This is a self-curing zinc-filled inorganic two-part basic zinc silicate complex

that is used to supply galvanic corrosion protection to steel surfaces in marine environments. The coating is insoluble in water and resistant to aggressive water and solvents. Polyurethane sheets located within the drywell gap are coated on both sides with an epoxy resin binder to prevent water ingress into the material. During the examination of containment that was completed during the seventh refueling outage (RF07), in the spring of 2000, visual inspections identified a small (0.02 x 0.04 x 0.093 inch deep) pit at the interface of an I-beam with the containment steel liner. The corrosion was attributed to a screw and uncoated washer that were in contact with an uncoated portion of the drywell shell in a beam seat area. The screw and washer were removed and the drywell shell in the area of the pit was coated in 2003. In addition, the Fermi 2 drywell in the vicinity of the sand pocket is 1.5 inches thick. The configuration of a Mark I containment has three primary areas susceptible to degradation. These are: (1) the moisture seal area where the concrete floor and containment steel liner meet; (2) the sand cushion area near the bottom of the drywell shell; and (3) the wet region inside the torus.

In RF07, during the first containment inspection interval, the moisture seal at the interface between the drywell concrete floor and the steel shell was removed. This was done to perform a detailed inspection of the liner in the seal area, repair areas of degradation in the seal, and as a preventive maintenance task. The inspection found no degradation to the drywell shell. The area was repainted and a new moisture seal was installed. In accordance with ASME Section XI, 2001 Edition with 2003 Addenda, the moisture barrier is inspected 100 percent each inspection period.

Fermi 2 has a sand pocket (cushion) region. Leakage is periodically monitored and has remained at zero since 1993. Fermi 2 personnel evaluated NRC Information Notice (IN) 2011-15, “Steel Containment Degradation and Associated License Renewal Aging Management.” (Reference 6.18) The history of sand pocket leakage and the design were evaluated. Based on no leakage being identified for 20 years and the protective coating on the drywell shell, accelerated actions were not deemed warranted, and the planning for a boroscope inspection and UT inspection of the shell in the vicinity of the sand pocket was initiated. Ultrasonic thickness measurements of the sand cushion region (approximately 50%) were performed in RF16 (Spring of 2014), with no loss of material identified.

In 2013, all four sand cushion drain lines were internally inspected with a boroscope. There were indications of sand in the bottom of three of the pipes and soft sand at the ends of the pipe. No water was present and the sand would have allowed a drain path through the pipe. In three cases, the boroscope reached the 90-degree elbow at the sand cushion; in the other case, boroscope could not reach the final foot of the pipe before the elbow because of the presence of sand. There were signs of corrosion in three of the pipes, showing that moisture had been present in the past.

Since 1995, all four drain lines are inspected for moisture on a quarterly basis. The quarterly inspections monitor the sand cushion area for moisture to ensure early detection of any condition conducive to corrosion.

The torus is inspected in alternate refueling outages, and the inspection is performed by certified personnel. NRC IN 2006-01, "Torus Cracking in a BWR Mark I Containment," (Reference 6.19) described a through-wall crack and its probable cause in the torus of a BWR Mark I containment. The cracking identified in the heat-affected zone at the HPCI turbine exhaust pipe torus penetration was most likely initiated by cyclic loading due to condensation oscillation during HPCI operation. The Fermi 2 HPCI design has a turbine exhaust pipe sparger that precludes this condition.

An inspection was performed in 2012, when 100 percent of the torus wetted and vapor space was inspected by qualified NDE inspectors. No pitting of the torus primary containment boundary was identified.

One 0.25 inch diameter pit has been identified in the torus wetted area during the history of the plant. The pit, a corrosion pit 0.0285 inches in depth, was identified under a coating blister in 2001. The depth of the pit left the remaining shell thickness well within design tolerances. The coating was repaired.

Coating condition continues to be monitored during inspections. In RF15, broken blisters, mechanical damage, and pinpoint rust areas were identified and repaired in the wetted areas of the torus. In the vapor region, all flaking paint was removed from the torus ring header, torus vacuum breaker valves, nitrogen supply lines, monorail rail, and torus walkway and handrail. Flaking or cracked coating was removed and protective coating was re-applied to the torus shell. As stated above, no pitting of the torus was identified during the 2012 (RF15) inspections.

For the leak tightness of seals, gaskets and bolted connections, DTE will continue to perform inspections approved by the NRC on these components as described in the containment inspection program. Seals and gaskets undergo alternative testing in accordance with 10 CFR 50, Appendix J, Type B. Bolted connections are tested per Appendix J and are subject to a general visual inspection once each containment inspection period. DTE also performs post-maintenance Appendix J testing following any repair or disassembly of a component with a seal, gasket, or bolted connection. These examinations will not be affected by the extension of the Type A test frequency.

The IWE examinations will be performed in accordance with the examination schedule per Table IWE-2500-1, which is shown in Table 5. There are currently no primary containment surface areas that require augmented examinations in accordance with ASME Section XI, Subsection IWE-1240.

Table 5: ASME CLASS MC EXAMINATIONS SCHEDULE FOR FERMI 2

EXAM CAT.	ITEM NO.	DESCRIPTION	EXAM METHOD	NUMBER OF COMPONENTS	INTERVAL REQMN'T FOR EXAM	NUMBER OF EXAMS FOR INTERVAL	REQ. OF EXAM DEFERRAL	NUMBER EXAMS 1ST PERIOD	NUMBER EXAMS 2ND PERIOD	NUMBER EXAMS 3RD PERIOD
E-A	**CONTAINMENT SURFACES**									
E-A	E1.10	Containment Vessel Pressure Retaining Boundary								
E-A	E1.11	Accessible Surface Areas	General Visual	1	100% Each Inspection Period	3	Deferral Not Permissible	1	1	1
E-A	E1.11B	Pressure Retaining Bolting (Note 1)	Visual, VT-3	101	100% Each Inspection Interval	101	Note 2	0	0	101
E-A	E1.12	Wetted Surfaces of Submerged Areas	Visual, VT-3 Note 3	1	100% Each Inspection Interval	1	Deferral Permissible	0	0	1
E-A	E1.20	Vent System – Accessible Surface Areas	Visual, VT-3 (Note 3)	1	100% Each Inspection Interval	1	Deferral Permissible	0	0	1
E-A	E1.30	Moisture Barrier	General Visual	1	100% Each Inspection Period		Deferral Not Permissible	1	1	1
E-C	**CONTAINMENT SURFACES REQUIRING AUGMENTED EXAMINATION**									
E-C	E4.10	Containment Surface Areas								
E-C	E4.11	Visible Surfaces	Visual, VT-1	0	100% Each Inspection Period	0	Deferral Not Permissible	0	0	0
E-C	E4.12	Surface Area Grind, Min Wall Thickness Locations	Volumetric UT	0	100% Each Inspection Period	0	Deferral Not Permissible	0	0	0

1) VT-3 examination of pressure retaining bolting is a requirement of 10 CFR 50.55a(b)(2)(ix)(G) and (H).
2) Per 10 CFR 50.55a(b)(2)(ix)(H), as an alternative to performing VT-3 examinations of containment bolted connections that are disassembled during the scheduled performance of Item E1.11, VT-3 examinations of containment bolted connections may be conducted whenever containment bolted connections are disassembled for any reason.VT-3 examination of pressure retaining bolting may be done while the component is disassembled during the General Visual of containment conducted each period.
3) VT-3 examination of Item Nos. E1.12 and E1.20 components and VT-1 examination of Item No E4.11 components is a requirement of 10 CFR 50.55a(b)(2)(ix)(G).

The last Fermi Type A test was completed in November 2007 during RF12. Based on a 15 year interval the next Fermi 2 Type A test would be required to be performed on or before November 7, 2022 (during Inspection Interval 3, Period 1). Based on Fermi 2 proposed outage dates, this would be required to be performed in RF21 (September 2021). Thus, three containment general visual examinations performed in accordance with the IWE program would take place prior to the 2022 Type A test (e.g., during Inspection Interval 2, Period 1, and Period 3, and then during Inspection Interval 3, Period 1). Fermi 2 is currently in the second period of the second containment inspection interval. Therefore, a 100% IWE inspection of accessible interior and exterior areas will be performed in the following inspection periods, as shown below:

Table 6: Schedule of 100% IWE Visual Inspections

Interval	Period	Outage(s)
1st Interval	3rd	RF12[3], RF13
2nd Interval	1st	RF14[1], RF15[2]
	2nd	RF16[1], RF17
	3rd	RF18[4], RF19
3rd Interval	1st	RF20[4], RF21[4] - ILRT
	2nd	RF22[5], RF23[6]
	3rd	RF24[5], RF25[6]

1: 100% visual inspection of accessible interior/exterior primary containment shell completed and torus exterior and torus interior vapor region completed.
2: 100% visual inspection of accessible interior torus wetted region completed.
3: 100% visual inspection of accessible interior/exterior primary containment shell completed and torus interior/exterior wetted and vapor region completed.
4: 100% visual inspection of accessible interior/exterior primary containment shell and torus interior/exterior wetted and vapor region required.
5: 100% visual inspection of accessible interior/exterior primary containment shell, torus exterior, and torus interior vapor region required.
6: 100% visual inspection of accessible interior torus wetted region required.

3.4 NRC Information Notice 92-20, "Inadequate Local Leak Rate Testing"

NRC IN 92-20, "Inadequate Local Leak Rate Testing," (Reference 6.20) discusses inadequate Type B LLRT of two-ply stainless steel bellows. Problems were identified with the testing of two-ply stainless steel bellows used on piping penetrations at some plants. Fermi 2 Deviation Event Report (DER) 92-0130 provided a response to IN 92-20. DTE determined, based on a review of the purchase specifications and discussion with the manufactures, that the bellows installed at Fermi 2 have a wire mesh between the plies that ensures an air gap for the adequate performance of Appendix J, Type B testing. In addition, no expansion bellow assemblies at Fermi 2 have leakage rate acceptance criteria greater than 1 scfh.

3.5 Plant-Specific Confirmatory Analysis

3.5.1 Methodology

An analysis has been performed to assess the risk impact of extending the Fermi 2 Type A test interval from 10 to 15 years, (Enclosure 4 to this letter). A simplified bounding analysis approach was used consistent with Appendix H of EPRI Report No. 1009325 (Reference 6.6), EPRI TR-104285 (Reference 6.12), NUREG-1493 (Reference 6.5) and the Calvert Cliffs liner corrosion analysis (Reference 6.21).

The analysis uses results from a Level 2 analysis of core damage scenarios from the current Fermi 2 probabilistic risk assessment (PRA) model and subsequent containment response resulting in various fission product release categories (including no or negligible release), and follows six general steps of this assessment. These steps include the following: determine the change in risk large early release frequency (LERF); determine the impact on Conditional Containment Failure Probability (CCFP); evaluate the risk impact of extending the ILRT interval to 15 years; quantify the baseline risk in terms of frequency of events for each of the eight containment release scenario types identified in the EPRI report; develop plant-specific person-rem per reactor year for each of the eight containment release scenario types form plant specific consequence analyses; and evaluate the sensitivity of the results to assumptions in the liner corrosion analysis, external events, and to the fractional contributions of increased large isolation failures (due to liner breach) to LERF.

3.5.2 Probabilistic Risk Assessment (PRA) Technical Adequacy

The technical adequacy of the Fermi 2 PRA is consistent with the requirements of Regulatory Guide 1.200, Revision 2, "An Approach for Determining the Technical Adequacy of Probabilistic Risk Assessment Results for Risk-Informed Activities," (Reference 6.22), as is relevant to this ILRT interval extension. Fermi 2 has Level 2 models that include internal events. Severe accident sequences have been developed from internally initiated events including internal floods. The sequences have been developed to determine the frequency for the radiological release to the environment. Information developed for FermiV10 of the PRA to support the Level 2 release categories is also used in this analysis.

The Fermi 2 PRA models are highly detailed and include a wide variety of initiating events, modeled systems, operator actions, and common cause events. The Fermi 2 model of record and supporting documentation has been maintained as a living program, with periodic updates to reflect the as-built, as-operated plant. The Fermi 2 Individual Plant Examination (IPE) and Individual Plant Examination of External Events (IPEEE) PRA models underwent NRC reviews, and updates to these models have been the subject of several assessments to establish the technical adequacy of the PRA. Documentation of Fermi 2 technical adequacy was submitted to the NRC in support of the approved license amendment request for adoption Risk Informed TS Initiative 5b (Reference 6.23). This previously submitted documentation of PRA technical adequacy is applicable for this Type A interval extension submittal.

3.5.3 Conclusion of Plant-Specific Risk Assessment Results

The Fermi 2 plant-specific results for extending the Type A test interval from the current 10 years to 15 years is summarized below from Enclosure 4, Section 7.

"Reg. Guide 1.174 provides guidance for determining the risk impact of plant specific changes to the licensing basis. Reg. Guide 1.174 defines very small changes in risk as resulting in increases of CDF below 1.0E-6/yr and increases in LERF below 1.0E-7 per reactor 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 three in ten years to one in fifteen years is very conservatively estimated as 1.27E-08/yr using the EPRI guidance as written. As such, the estimated change in LERF is determined to be "very small" using the acceptance guidelines of Reg. Guide 1.174.

Regulatory Guide 1.174 also states that when the calculated increase in LERF is in the range of 1.0E-06 per reactor year to 1.0E-07 per reactor year, applications will be considered only if it can be reasonably shown that the total LERF is less than 1.0E-05 per reactor year. An additional assessment of the impact from external events was also made. In this case, the total LERF increase was conservatively estimated (with an external event multiplier of 15) as 1.90E-07 for Fermi 2 (the baseline total LERF for this case is 7.88E-06/yr). This is well below the RG 1.174 acceptance criteria for total LERF of 1.0E-05.

The change in Type A test frequency to once-per-fifteen-years, measured as an increase to the total integrated plant risk for those accident sequences influenced by Type A testing, is 1.14E-4 person-rem/yr (a 0.00184% increase). EPRI Report No. 1009325, Revision 2-A 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. Moreover, the risk impact when compared to other severe accident risks is negligible.

The increase in the CCFP from the three in ten year interval to one in fifteen year interval is 0.73%. EPRI Report No. 1009325, Revision 2-A states that increases in CCFP of less than or equal to 1.5 percentage points are very small. Therefore this increase judged to be very small.

A sensitivity case was performed which examined the effect of containment overpressure on CDF and LERF; it should be noted that containment overpressure is not credited in the Fermi design basis. Substantial conservatism was included in the sensitivity to ensure that it was bounding. The increase in CDF for this sensitivity was 5.25E-08/yr and the increase in LERF was 3.54E-08/yr. The estimated change in both CDF and LERF was determined to be "very small" using the acceptance guidelines of Reg. Guide 1.174. The estimated change in both CDF and LERF was determined to be "very small" using the acceptance guidelines in Regulatory Guide 1.174. This sensitivity was utilized to justify the screening of the CDF impact from further consideration.

Therefore, increasing the ILRT interval to once per 15 years is considered to be insignificant, since it represents a very small change to the Fermi 2 risk profile."

3.6 Conclusion

NEI 94-01, Revision 3-A, dated July 2012, and the conditions and limitations specified in NEI 94-01, Revision 2-A, dated October 2008, describe an NRC accepted approach for implementing the performance-based requirements of Appendix J, Option B. It incorporates the regulatory positions stated in Regulatory Guide 1.163 and includes provisions for extending Type A test intervals to 15 years and Type C test intervals to 75 months. NEI 94-01, Revision 3-A, delineates a performance-based approach for determining Type A, Type B, and Type C containment leakage rate surveillance test frequencies. DTE is adopting the guidance of NEI 94-01, Revision 3-A, and the conditions and limitations specified in NEI 94-01, Revision 2-A, for use in Fermi 2's 10 CFR Part 50, Appendix J testing program plan.

Based on the previous ILRT tests conducted at Fermi 2, the permanent extension of the containment ILRT interval from one in 10 years to one in 15 years represents minimal risk to increased leakage. This risk is minimized by continued Type B and Type C testing performed in accordance with Option B of 10 CFR Part 50, Appendix J and the overlapping inspection activities performed as part of the following Fermi 2 inspection programs:

- Containment Inservice Inspection Program (IWE)
- Containment Inspections per TS SR 3.6.1.1.1

This is confirmed by the risk analysis provided in Enclosure 4. The findings of the risk assessment confirm the general findings of previous studies, on a plant-specific basis, that extending the ILRT interval from 10 to 15 years results in a very small change to the Fermi 2 risk profile.

4.0 Regulatory Evaluation

4.1 Applicable Regulatory Requirements/Criteria

10 CFR 50.54(o) requires primary reactor containments for water-cooled power reactors to be subject to the requirements of Appendix J to 10 CFR Part 50, "Leakage Rate Testing of Containment of Water-Cooled Nuclear Power Plants." Appendix J, Option B, requires that licensees' primary reactor containment meet the leakage rate requirements as delineated in Appendix J. Appendix J specifies containment leakage testing requirements, including the types required to ensure the leak-tight integrity of the primary reactor containment and systems and components which penetrate the containment. It also discusses leakage rate acceptance criteria, test methodology, frequency of testing, and reporting requirements for each type of test. Regulatory Guide 1.163 (Reference 6.2) provides a method acceptable to the NRC for implementing the performance-based option (Option B) of 10 CFR Part 50, Appendix J. The regulatory positions stated in RG 1.163 as modified by the NRC SERs, dated June 25, 2008, and June 8, 2012, for NEI 94-01, Revisions 2-A and 3-A, respectively, are incorporated in NEI 94-01, Revision 3-A. NEI 94-01, Revisions 2-A and 3-A, each incorporate the applicable NRC SER, including limitations and conditions delineated in Section 4.0 of each SER (References 6.3 and 6.1, respectively). Subject to the limitations and conditions in the applicable SER, each revision of NEI 94-01 was determined to provide an acceptable approach for implementing the optional performance-based requirements of Option B to 10 CFR Part 50, Appendix J. NEI 94-01, Revision 3-A, includes guidance for extending the ILRT Type A test interval to 15 years and the Type C LLRT test interval up to 75 months. Any applicant may reference NEI 94-01, Revision 3-A, as modified by the associated SER and approved by the NRC, and the conditions and limitations specified in NEI 94-01, Revision 2-A, in a licensing action to satisfy the requirements of Option B to 10 CFR Part 50, Appendix J.

The proposed license amendment would revise Fermi 2 TS 5.5.12 "Primary Containment Leakage Rate Testing Program," item a, by changing the wording to indicate that the program shall be in accordance with NEI 94-01, Revision 3-A, and the conditions and limitations specified in NEI 94-01, Revision 2-A, instead of Regulatory Guide 1.163.

The proposed license amendment also proposes two administrative changes. First, a historical one-time extension to the Type A test interval in TS 5.5.12, item a, will be removed. Second, an exemption to Appendix J testing of containment airlocks will be removed from Operating License Provision D. Amendment 108 (Reference 6.8) eliminated the need for this exemption, but it was not removed from the Operating License.

4.2 Precedent

This request is similar in nature to the following license amendments to extend the Type A Test Frequency to 15 years, and to extend Type C test intervals to 75 months, as previously authorized by the NRC:

- Nine Mile Point Nuclear Station, Unit 2 (Reference 6.24)
- Arkansas Nuclear One, Unit 2 (Reference 6.25)
- Peach Bottom Atomic Power Stations, Units 2 and 3 (Reference 6.26)
- Donald C. Cook Nuclear Plants, Units 1 and 2 (Reference 6.27)

4.3 No Significant Hazards Consideration

DTE has evaluated whether or not a significant hazards consideration is involved with the proposed amendment for extending the ILRT Type A interval from 10 to 15 years and extending the Type C test interval from 60 to 75 months by focusing on the three standards set forth in 10 CFR 50.92, "Issuance of Amendment," as discussed below:

1. Does the proposed amendment involve a significant increase in the probability or consequences of an accident previously evaluated?

   Response: No.

   The proposed amendment to the TS involves the extension of Fermi 2 Type A containment test interval to 15 years and the extension of the Type C test interval to 75 months. The current Type A test interval of 10 years would be extended on a permanent basis to no longer than 15 years from the last Type A test. The current Type C test interval of 60 months for selected components would be extended on a performance basis to no longer than 75 months. Extensions of up to nine months (total maximum interval of 84 months for Type C tests) are permissible only for non-routine emergent conditions. The proposed amendment does not involve either a physical change to the plant or a change in the manner in which the plant is operated or controlled. The primary containment is designed to provide an essentially leak tight barrier against the uncontrolled release of radioactivity to the environment for postulated accidents. As such, the containment and the testing requirements invoked to periodically demonstrate the integrity of the containment exist to ensure the plant's ability to mitigate the consequences of an accident, and do not involve any accident precursors or initiators. RG 1.174 (Reference 6.4) provides guidance for determining the risk impact of plant-specific changes to the licensing basis. RG 1.174 defines very small changes in risk as resulting in increases of CDF below 1.0E-06/yr and increases in LERF below 1.0E-07/yr. Since the ILRT does not impact CDF, the relevant criterion is LERF. The increase in LERF resulting from a change in the Type A ILRT test interval from three in ten years to one in fifteen years is very conservatively estimated as 1.27E-08/yr using the EPRI guidance as written. As such, the estimated change in LERF is determined to be "very small" using the acceptance guidelines of RG 1.174.

   RG 1.174 also states that when the calculated increase in LERF is in the range of 1.0E-06 per reactor year to 1.0E-07 per reactor year, applications will be considered only if it can be reasonably shown that the total LERF is less than 1.0E-05 per reactor year. An additional assessment of the impact from external events was also made. In this case, the total LERF increase was conservatively estimated (with an external event multiplier of 15) as 1.90E-07

for Fermi 2 (the baseline total LERF for this case is 7.88E- 06/yr). This is well below the RG 1.174 acceptance criteria for total LERF of 1.0E-05.

The change in Type A test frequency to once per 15 years, measured as an increase to the total integrated plant risk for those accident sequences influenced by Type A testing, is 1.14E-4 person-rem/yr (a 0.00184% increase). EPRI Report No. 1009325, Revision 2-A, 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. Moreover, the risk impact when compared to other severe accident risks is negligible.

The increase in the CCFP from the three in 10 years interval to one in 15 years interval is 0.73%. EPRI Report No. 1009325, Revision 2-A, states that increases in CCFP of less than or equal to 1.5 percentage points are very small. Therefore this increase judged to be very small.

The other two changes, to TS 5.5.12, item a, and Operating License, Provision D, are administrative in nature to remove old text that is no longer applicable.

Therefore, the proposed change does not involve a significant increase in the probability or consequences of an accident previously evaluated.

2. Does the proposed change create the possibility of a new or different kind of accident from any accident previously evaluated?

   Response: No.

   The proposed amendment to the TS involves the extension of the Fermi 2 Type A containment test interval to 15 years and the extension of the Type C test interval to 75 months. The containment and the testing requirements to periodically demonstrate the integrity of the containment exist to ensure the plant's ability to mitigate the consequences of an accident and do not involve any accident precursors or initiators. The proposed change does not involve a physical change to the plant (e.g., no new or different type of equipment will be installed) or a change to the manner in which the plant is operated or controlled.

   The other two changes to TS 5.5.12, item a, and Operating License, Provision D, are administrative in nature to remove old text that is no longer needed. Therefore, these changes have no impact on the probability or consequences of an accident previously evaluated.

   Therefore, the proposed change does not create the possibility of a new or different kind of accident from any previously evaluated.

3. Does the proposed change involve a significant reduction in a margin of safety?

   Response: No.

   The proposed amendment to TS 5.5.12 involves the extension of the Fermi 2 Type A containment test interval to 15 years and the extension of the Type C test interval to 75 months for selected components. This amendment does not alter the manner in which safety limits, limiting safety system set points, or limiting conditions for operation are determined. The specific requirements and conditions of the TS Containment Leak Rate Testing Program exist to ensure that the degree of containment structural integrity and leak-tightness that is considered in the plant safety analysis is maintained. The overall containment leak rate limit specified by TS is maintained.

   The proposed surveillance interval extension is bounded by the 15 year ILRT interval and the 75 month Type C test interval currently authorized within NEI 94-01, Revision 3-A. Industry experience supports the conclusion that Type B and Type C testing detects a large percentage of containment leakage paths and the the percentage of containment leakage paths that are detected only by Type A testing is small. The containment inspections preformed in accordance with ASME Section XI, Maintenance Rule, and TS serve to provide a high degree of assurance that the containment would not degrade in a manner that is detectable only by Type A testing. The combination of these factors ensures that the margin of safety in the plant safety analysis is maintained. The design, operation, testing methods, and acceptance criteria for Type A, Type B, and Type C containment leakage tests specified in applicable codes and standards would continue to be met with the acceptance of this proposed change since these are not affected by the changes to the Type A and Type C test intervals.

   The other two changes to TS 5.5.12, item a, and Operating License, Provision D, are administrative in nature to remove old text that is no longer needed. Therefore, these changes have no impact on the probability or consequences of an accident previously evaluated.

   Therefore, the proposed change does not involve a significant reduction in a margin of safety.

   Based on the above, DTE concludes that the proposed change presents no significant hazards consideration under the standards set forth in 10 CFR 50.92(c), and, accordingly, a finding of no significant hazards consideration is justified.

4.4 Conclusions

In conclusion, based on the considerations discussed above, (1) there is reasonable assurance that the health and safety of the public will not be endangered by operation in the proposed manner, (2) such activities will be conducted in compliance with the Commission's regulations, and (3)

the issuance of the amendment will not be inimical to the common defense and security or to the health and safety of the public.

5.0 Environmental Consideration

A review has determined that the proposed amendment would change surveillance requirements regarding leak rate testing of the primary containment and make two administrative changes to the Fermi 2 TS. The proposed amendment does not involve: (i) a significant hazards consideration, (ii) a significant change in the types or significant increase in the amounts of any effluent that may be released offsite, or (iii) a significant increase in individual or cumulative occupational radiation exposure. Accordingly, the proposed amendment meets the eligibility criterion for categorical exclusion set forth in 10 CFR 51.22(c)(9). Therefore, pursuant to 10 CFR 51.22(b), no environmental impact statement or environmental assessment need be prepared in connection with the proposed amendment.

6.0 References

6.1 NEI Topical Report 94-01, Revision 3-A, “Industry Guideline for Implementing Performance-Based Option of 10 CFR Part 50, Appendix J,” July 2012 (ML12221A202)
6.2 NRC Regulatory Guide 1.163, Revision 0, “Performance Based Containment Leak-Test Program,” September 1995 (ML003740058)
6.3 NEI Topical Report 94-01, Revision 2-A, “Industry Guideline for Implementing Performance-Based Option of 10 CFR Part 50, Appendix J,” October 2008 (ML100620847)
6.4 NRC Regulatory Guide 1.174, Revision 0, “An Approach for Using Probabilistic Risk Assessment in Risk-Informed Decisions on Plant-Specific Changes to the Licensing Basis,” July 1998 (ML003740133)
6.5 NRC NUREG-1493, “Performance-Based Containment Leak-Test Program,” September 1995 (9510200161)
6.6 EPRI Report No. 1009325, Revision 2, “Risk Impact Assessment of Extended Integrated Leak Rate Testing Intervals,” August 2007 (ML072970208)
6.7 DTE Letter, NRC-14-0028, “Fermi 2 License Renewal Application,” dated April 24, 2014 (ML14121A532)
6.8 NRC Letter, “Fermi-2 – Issuance of Amendment Re: Implementation of 10 CFR Part 50 Appendix J Option B,” August 8, 1996 (ML020730597)
6.9 NRC NUREG-0661, “Safety Evaluation Report – Mark I Containment Long-Term Program, resolution of Generic Technical Activity A-7,” July 1980 (ML11203A031)
6.10 GE Letter to NRC, "Mark I Containment Program - General Electric Report, NEDO-21888, Mark I Containment Program Load Definition Report," dated December 28, 1978 (proprietary)
6.11 GE Licensing Topical Report NEDC-31897P-1, “Generic Guidelines for General Electric Boiling Water Reactor Power Uprate,” June 1991 (proprietary)
6.12 EPRI Report No. 104285, “Risk Impact Assessment of Revised Containment Leak Rate Testing Intervals,” 1994

6.13 American Nuclear Society, ANSI/ANS-56.8, "Containment System Leakage Testing Requirements," 2002 (not public)
6.14 NRC Letter, "Fermi 2 - Correction to Amendment No. 153 Regarding a One-Time Deferral of the Primary Containment Integrated Leak Rate Test," April 1, 2003 (ML030930443)
6.15 NEI Topical Report 94-01, Revision 0, "Industry Guideline for Implementing Performance-Based Option of 10 CFR Part 50, Appendix J," July 21, 1995 (ML11327A025)
6.16 DTE Letter, NRC-06-0037, "Licensee Event Report No. 2006-001, 'Excessive Feedwater Check Valve Leakage at Containment Penetration,'" dated May 24, 2006 (ML061520405)
6.17 DTE Letter, NRC-07-0057, "Licensee Event Report No. 2007-001, 'Excessive Feedwater Check Valve Leakage at Containment Penetration,'" dated December 6, 2007 (ML073460442)
6.18 NRC Information Notice 2011-15, "Steel Containment Degradation and Associated License Renewal Aging Management Issues," August 1, 2011 (ML111460369)
6.19 NRC Information Notice 2006-01, "Torus Cracking in a BWR Mark I Containment," January 12, 2006 (ML053060311)
6.20 NRC Information Notice 92-20, "Inadequate Local Leak Rate Testing," March 3, 1992 (ML031200473)
6.21 Calvert Cliffs Nuclear Power Plant Letter to NRC, "Response to Request for Additional Information Concerning the License Amendment Request for a One-Time Integrated Leakage Rate Test Extension," March 27, 2002 (ML020920100)
6.22 NRC Regulatory Guide 1.200, Revision 2, "An Approach for Determining the Technical Adequacy of Probabilistic Risk Assessment Results for Risk-Informed Activities," March 2009 (ML090410014)
6.23 NRC Letter, "Fermi 2 – Issuance of Amendment Re: Revise Technical Specifications by Relocating Surveillance Frequencies to Licensee Control in Accordance with Technical Specification Task Force Traveler 425, Revision 3," July 14, 2015 (ML151558416)
6.24 NRC Letter, "Nine Mile Point Nuclear Station, Unit No. 2 – Issuance of Amendment Re: Extension of Primary Containment Integrated Leakage Rate Testing Interval," March 30, 2010 (ML100730032)
6.25 NRC Letter, "Arkansas Nuclear One, Unit No. 2 – Issuance of Amendment Re: Technical Specification Change to Extend Type A Test Frequency to 15 Years," April 7, 2011 (ML110800034)
6.26 NRC Letter, "Peach Bottom Atomic Power Station, Units 2 and 3 – Issuance of Amendments Re: Extensions of Type A and Type C Leak Rate Test Frequencies," September 8, 2015 (ML15196A559)
6.27 NRC Letter, "Donald C. Cook Nuclear Plant, Units 1 and 2 – Issuance of Amendments Re: Containment Leakage Rate Testing Program," March 30, 2015 (ML15072A264)

Enclosure 2 to
NRC-16-0006

Fermi 2 NRC Docket No. 50-341
Operating License No. NPF-43

Marked-up Pages of Existing Fermi 2 Operating License and TS

D. Exemptions from certain requirements of Appendices E and J to 10 CFR Part 50, are described in supplements to the SER. These include: (a) an exemption from the requirement of Section IV.F of Appendix E that a full participation emergency planning exercise be conducted within one year before issuance of the first
and operating license for full power and prior to operation above five percent of rated power (Section 13.3 of SSER #6); (b) an exemption from the requirement of Paragraph III.C.2(b) of Appendix J, the testing of the main steam isolation valves at the peak calculated containment pressure associated with the design basis accident (Section 6.2.7 of SSER #5)~~; and (c) an exemption from the requirement of Paragraph III.D.2(b)(ii) of Appendix J, the testing of containment air locks at times when containment integrity is not required (Section 6.2.7 of SSER #5)~~. These exemptions are authorized by law and will not endanger life or property or the common defense and security and are otherwise in the public interest. Therefore, these exemptions are hereby granted pursuant to 10 CFR 50.12. With the granting of these exemptions, the facility will operate, to the extent authorized herein, in conformity with the application, as amended, the provisions of the Act, and the rules and regulations of the Commission.

E. The licensee shall fully implement and maintain in effect all provisions of the Commission-approved physical security, training and qualification, and safeguards contingency plans including amendments made pursuant to provisions of the Miscellaneous Amendments and Search Requirements revisions to 10 CFR 73.55 (51 FR 27817 and 27822) and to the authority of 10 CFR 50.90 and 10 CFR 50.54(p). The plans, which contain Safeguards Information protected under 10 CFR 73.21, are entitled: "Fermi 2 Physical Security Plan, Security Training and Qualification Plan, and Safeguards Contingency Plan" submitted by letter dated September 9, 2004, and supplemented on October 7, 2004, and October 14, 2004, November 18, 2005, and May 18, 2006. The licensee shall fully implement and maintain in effect all provisions of the Commission-approved cyber security plan (CSP), including changes made pursuant to the authority of 10 CFR 50.90 and 10 CFR 50.54(p). The Fermi 2 CSP was approved by License Amendment No. 185, as supplemented by License Amendment 200.

F. Deleted

G. The licensees shall have and maintain financial protection of such type and in such amounts as the Commission shall require in accordance with Section 170 of the Atomic Energy Act of 1954, as amended, to cover public liability claims.

5.5 Programs and Manuals

5.5.11 Safety Function Determination Program (SFDP) (continued)

The SFDP identifies where a loss of safety function exists. If a loss of safety function is determined to exist by this program, the appropriate Conditions and Required Actions of the LCO in which the loss of safety function exists are required to be entered.

5.5.12 Primary Containment Leakage Rate Testing Program

a. A program shall be established to implement the leakage rate testing of the primary containment as required by 10 CFR 50.54(o) and 10 CFR 50, Appendix J, Option B, as modified by approved exemptions. This program shall be in accordance with ~~the guidelines contained in Regulatory Guide 1.163, "Performance-Based Containment Leak-Test Program," dated September, 1995, with the exception of approved exemptions to 10 CFR 50, Appendix J, and as modified by the following exception to NEI 94-01, Rev. 0, "Industry Guideline for Implementing Performance-Based Option of 10 CFR 50, Appendix J":~~

Nuclear Energy Institute (NEI) 94-01, Revision 3-A, "Industry Guideline for Implementing Performance-Based Option of 10 CFR Part 50, Appendix J," dated July 2012, and the limitations and conditions specified in NEI 94-01, Revision 2-A, dated October 2008.

~~The first Type A test after the October 1992 test shall be performed no later than October 2007.~~

b. The peak calculated containment internal pressure for the design basis loss of coolant accident, P_a, is 56.5 psig.

c. The maximum allowable containment leakage rate L_a, at P_a, shall be 0.5% of containment air weight per day.

d. Leakage Rate acceptance criteria are:

1. Containment leakage rate acceptance criterion is $\leq 1.0\ L_a$. During the first unit startup following testing in accordance with this program, the leakage rate acceptance criteria are $\leq 0.60\ L_a$ for the required Type B and C tests and $\leq 0.75\ L_a$ for Type A tests.

2. Air lock testing acceptance criteria are:

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

ii) For each door, leakage rate is ≤ 5 scf per hour when the gap between the door seals is pressurized to $\geq P_a$.

(continued)

Enclosure 3 to
NRC-16-0006

Fermi 2 NRC Docket No. 50-341
Operating License No. NPF-43

Clean Pages of Fermi 2 Operating License and TS
with Changes Incorporated

D. Exemptions from certain requirements of Appendices E and J to 10 CFR Part 50, are described in supplements to the SER. These include: (a) an exemption from the requirement of Section IV.F of Appendix E that a full participation emergency planning exercise be conducted within one year before issuance of the first operating license for full power and prior to operation above five percent of rated power (Section 13.3 of SSER #6); and (b) an exemption from the requirement of Paragraph III.C.2(b) of Appendix J, the testing of the main steam isolation valves at the peak calculated containment pressure associated with the design basis accident (Section 6.2.7 of SSER #5). These exemptions are authorized by law and will not endanger life or property or the common defense and security and are otherwise in the public interest. Therefore, these exemptions are hereby granted pursuant to 10 CFR 50.12. With the granting of these exemptions, the facility will operate, to the extent authorized herein, in conformity with the application, as amended, the provisions of the Act, and the rules and regulations of the Commission.

E. The licensee shall fully implement and maintain in effect all provisions of the Commission-approved physical security, training and qualification, and safeguards contingency plans including amendments made pursuant to provisions of the Miscellaneous Amendments and Search Requirements revisions to 10 CFR 73.55 (51 FR 27817 and 27822) and to the authority of 10 CFR 50.90 and 10 CFR 50.54(p). The plans, which contain Safeguards Information protected under 10 CFR 73.21, are entitled: "Fermi 2 Physical Security Plan, Security Training and Qualification Plan, and Safeguards Contingency Plan" submitted by letter dated September 9, 2004, and supplemented on October 7, 2004, and October 14, 2004, November 18, 2005, and May 18, 2006. The licensee shall fully implement and maintain in effect all provisions of the Commission-approved cyber security plan (CSP), including changes made pursuant to the authority of 10 CFR 50.90 and 10 CFR 50.54(p). The Fermi 2 CSP was approved by License Amendment No. 185, as supplemented by License Amendment 200.

F. Deleted

G. The licensees shall have and maintain financial protection of such type and in such amounts as the Commission shall require in accordance with Section 170 of the Atomic Energy Act of 1954, as amended, to cover public liability claims.

5.5 Programs and Manuals

5.5.11 Safety Function Determination Program (SFDP) (continued)

The SFDP identifies where a loss of safety function exists. If a loss of safety function is determined to exist by this program, the appropriate Conditions and Required Actions of the LCO in which the loss of safety function exists are required to be entered.

5.5.12 Primary Containment Leakage Rate Testing Program

a. A program shall be established to implement the leakage rate testing of the primary containment as required by 10 CFR 50.54(o) and 10 CFR 50, Appendix J, Option B, as modified by approved exemptions. This program shall be in accordance with Nuclear Energy Institute (NEI) 94-01, Revision 3-A, "Industry Guideline for Implementing Performance-Based Option of 10 CFR Part 50, Appendix J," dated July 2012, and the limitations and conditions specified in NEI 94-01, Revision 2-A, dated October 2008.

b. The peak calculated containment internal pressure for the design basis loss of coolant accident, P_a, is 56.5 psig.

c. The maximum allowable containment leakage rate L_a, at P_a, shall be 0.5% of containment air weight per day.

d. Leakage Rate acceptance criteria are:

   1. Containment leakage rate acceptance criterion is $\leq 1.0\ L_a$. During the first unit startup following testing in accordance with this program, the leakage rate acceptance criteria are $\leq 0.60\ L_a$ for the required Type B and C tests and $\leq 0.75\ L_a$ for Type A tests.

   2. Air lock testing acceptance criteria are:

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

      ii) For each door, leakage rate is ≤ 5 scf per hour when the gap between the door seals is pressurized to $\geq P_a$.

(continued)

Enclosure 4 to
NRC-16-0006

Fermi 2 NRC Docket No. 50-341
Operating License No. NPF-43

Probabilistic Risk Assessment Report

MARACOR

A division of ENERCON

Evaluation of Risk Significance of Permanent ILRT Extension

Fermi 2 Nuclear Power Plant

Revision 0

Prepared by: Joseph Lavelline 12/17/15

Joseph Lavelline, Maracor

Reviewed by: Nathan DeKett 12/17/15

Nathan DeKett, DTE Energy

Forward

The following analysis assesses the impact of a permanent extension of the Integrated Leak Rate Test at Fermi 2 from 10 to 15 years. The information presented in and the format of this analysis is derived from the template contained in Appendix H of EPRI 1009325, Revision 2-A (Reference 26). The main purpose of the template in Reference 26 is to illustrate the type of information that should be included in a plant-specific confirmation of risk impact associated with the extension of ILRT intervals.

Table of Contents

List of Tables

List of Figures

1. <u>Analysis Overview</u>

1.0 Purpose

The purpose of this analysis is to provide a risk assessment of extending the currently allowed containment Type A integrated leak rate test (ILRT) to a permanent interval of fifteen years. The extension would allow for substantial cost savings as the ILRT could be deferred for additional scheduled refueling outages for the Fermi 2 Nuclear Power Plant. The risk assessment follows the guidelines from:

1. NEI 94-01, Revision 2 (Reference 1), the methodology used in EPRI TR-104285 (Reference 2).
2. The NEI document *Interim Guidance for Performing Risk Impact Assessments In Support of One-Time Extensions for Containment Integrated Leakage Rate Test Surveillance Intervals* (Reference 3).
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.
4. Risk insights in support of a request for a plant's licensing basis as outlined in Regulatory Guide (RG) 1.174 (Reference 4).
5. The methodology used for Calvert Cliffs to estimate the likelihood and risk implications of corrosion-induced leakage of steel liners going undetected during the extended test interval (Reference 5).
6. The methodology used in EPRI 1009325, Revision 2-A (Reference 26).

1.1 Background

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

The basis for the current 10-year test interval is provided in Section 11.0 of NEI 94-01, Revision 0, and was established in 1995 during development of the performance-based Option B to Appendix J. Section 11.0 of NEI 94-01 (Reference 1) states that NUREG-1493, "Performance-Based Containment Leak Test Program," September 1995 (Reference 6), provides the technical basis to support rulemaking to revise leakage rate testing requirements contained in Option B to Appendix J. The basis consisted of qualitative and quantitative assessments of the risk impact (in terms of increased public dose) associated with a range of extended leakage rate test intervals. To supplement the NRC's rulemaking basis, NEI undertook a similar study. The results of that study are documented in Electric Power Research Institute (EPRI) Research Project Report TR- 104285, "Risk Impact Assessment of Revised Containment Leak Rate Testing Intervals." The NRC report on performance-based leak testing, NUREG-1493, analyzed the effects of containment leakage on the health and safety of the public and the benefits realized from the

containment leak rate testing. In that analysis, it was determined that for a representative PWR plant (i.e., Surry), that containment isolation failures contribute less than 0.1 percent to the latent risks from reactor accidents. Consequently, it is desirable to show that extending the ILRT interval will not lead to a substantial increase in risk from containment isolation failures for the Fermi 2 Nuclear Power Plant

The Guidance provided in Appendix H of EPRI Report No. 1009325, Revision 2-A, *Risk Impact Assessment of Extended Integrated Leak Rate Testing Intervals* (Reference 26) for performing risk impact assessments in support of ILRT extensions 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.

1.2 Acceptance Criteria

The acceptance guidelines in RG 1.174 are used to assess the acceptability of this one-time 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, an assessment of the increase in the conditional containment failure probability (CCFP) helps to ensure that the defense-in-depth philosophy is maintained.

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% (percentage point) is assumed to be small.

In addition, the total annual risk (person rem/yr population dose) is examined to demonstrate the relative change in this parameter. While no acceptance guidelines for these additional figures of merit are published, examinations of NUREG-1493 and Safety Evaluation Reports (SER) for one-time interval extension (summarized in Appendix G) indicate a range of incremental increases in population dose that have been accepted by the NRC (Reference 11). The range of incremental population dose increases is from ≤.0.01 to 0.2 person-rem/yr and/or 0.002 to 0.46% of the total accident dose. 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.

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

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

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

Moreover, it is noted that the CLIIP notice associated with promulgation of NEI 94-01, Revision 2 will indicate that the CLIIP only applies to those plants that do not credit

overpressure for ECCS pump operation and a traditional license amendment require is required in those cases where overpressure is credited.

2. Methodology

A simplified bounding analysis approach consistent with the EPRI approach is used for evaluating the change in risk associated with increasing the test interval to fifteen years. The approach is consistent with that presented in Appendix H of EPRI Report No. 1009325, Revision 2, *Risk Impact Assessment of Extended Integrated Leak Rate Testing Intervals* (Reference 26), EPRI TR-104285 (Reference 2), NUREG-1493 (Reference 6) and the Calvert Cliffs liner corrosion analysis (Reference 5). The analysis uses results from a Level 2 analysis of core damage scenarios from the current Fermi 2 PRA model and subsequent containment response resulting in various fission product release categories (including no or negligible release).

The six general steps of this assessment are as follows:

1. Quantify the baseline risk in terms of the frequency of events (per reactor year) for each of the eight containment release scenario types identified in the EPRI report.
2. Develop plant-specific person-rem (population dose) per reactor year for each of the eight containment release scenario types from plant specific consequence analyses.
3. Evaluate the risk impact (i.e., the change in containment release scenario type frequency and population dose) of extending the ILRT interval to fifteen years.
4. Determine the change in risk in terms of Large Early Release Frequency (LERF) in accordance with RG 1.174 (Reference 4) and compare with the acceptance guidelines of RG 1.174.
5. Determine the impact on the Conditional Containment Failure Probability (CCFP)
6. Evaluate the sensitivity of the results to assumptions in the liner corrosion analysis, external events and to the fractional contribution of increased large isolation failures (due to liner breach) to LERF.

This approach is based on the information and approaches contained in the previously mentioned studies. Furthermore,

- Consistent with the other industry containment leak risk assessments, the Fermi 2 assessment uses LERF and delta-LERF in accordance with the risk acceptance guidance of RG 1.174. Changes in population dose and conditional containment failure probability are also considered to show that defense-in-depth and the balance of prevention and mitigation is preserved.
- If containment overpressure is credited in the ECCS recirculation analysis and is included in the assessment an additional figure of merit is core damage frequency (CDF) to ensure that the guidelines from RG 1.174 are met.
- This evaluation for Fermi 2 uses ground rules and methods to calculate changes in risk metrics that are similar to those used in Appendix H of EPRI Report No. 1009325, Revision 2, *Risk Impact Assessment of Extended Integrated Leak Rate Testing Intervals* (Reference 26).

3. Calculation Ground Rules and Assumptions

The following ground rules and assumptions are used in the analysis (it should be noted that many of these ground rules were cited explicitly in Reference 26):

1. The technical adequacy of the Fermi 2 PRA is consistent with the requirements of Regulatory Guide 1.200 as is relevant to this ILRT interval extension.
2. The Fermi 2 Level 1 and Level 2 internal events PSA models provide representative results.
3. It is appropriate to use the Fermi 2 internal events PSA model as a gauge to effectively describe the risk change attributable to the ILRT extension. It is reasonable to assume that the impact from the ILRT extension (with respect to percent increases in population dose) will not substantially differ if fire and seismic events were to be included in the calculations.
4. Dose results for the containment failures modeled in the PSA can be characterized by information provided for the Severe Accident Management Alternative (SAMA) analysis in the *Fermi 2 WinMACCS Assessment of Severe Accident Consequences* (Reference 27) and also in the SAMA portion of the Fermi 2 License Renewal Application (Reference 28).
5. Accident classes describing radionuclide release end states are defined consistent with EPRI methodology (Reference 2) and are summarized in Section 4.2.
6. The representative containment leakage for Class 1 sequences is 1La Class 3 accounts for increased leakage due to Type A inspection failures.
7. The representative containment leakage for Class 3a sequences is 10La based on the previously approved methodology performed for Indian Point Unit 3 (References 8 and 9).
8. The representative containment leakage for Class 3b sequences is 100La based on the guidance provided in EPRI Report No. 1009325, Revision 2.
9. The Class 3b can be very conservatively categorized as LERF based on the previously approved methodology (References 8 and 9).
10. The impact on population doses from containment bypass scenarios is not altered by the proposed ILRT extension, but is accounted for in the EPRI methodology as a separate entry for comparison purposes. Since the containment bypass contribution to population dose is fixed, no changes on the conclusions from this analysis will result from this separate categorization.
11. The reduction in ILRT frequency does not impact the reliability of containment isolation valves to close in response to a containment isolation signal.
12. All quantifications for this analysis were performed at a 1.0E-12/yr truncation limit.
13. The "with maintenance" Fermi 2 Model of Record (MOR), Fermi V10 (Reference 17) was used for all quantitative results.
14. The random containment large isolation failure probability for large valves (8.60E-4) is assumed to equal the Fermi 2 Bayesian-updated probability for a safety-related motor-operated valve failing to close on demand found in Reference 18 (*Fermi 2 Component Data Notebook, Vol. 1, Revision 1*). The assumption is consistent with the wording in Section 4.3 of Reference 26, which states, "*Class 2 sequences: This group consists of all core damage accident progression bins for which a pre-existing leakage due to failure to isolate the containment occurs. These sequences are dominated by failure to close of large (>2 inches [5.1 cm] in diameter) containment isolation valves.*"

4. <u>Calculation Inputs</u>

This section summarizes the general resources available as input (Section 4.1) and the plant specific resources required (Section 4.2).

4.0 Industry 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, Appendix H (Reference 26)

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 Fermi 2. 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. The contents of the above resources are briefly discussed below.

<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.

NUREG/CR-4220 (Reference 11)

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

NUREG-1273 (Reference 12)

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

NUREG/CR-4330 (Reference 13)

NUREG/CR-4330 is a study that examined the risk impacts associated with increasing the allowable containment leakage rates. The details of this report have no direct impact on the modeling approach of the ILRT test interval extension, as NUREG/CR-4330 focuses on leakage rate and the ILRT test interval extension study focuses on the frequency of testing intervals. However, the general conclusions of NUREG/CR-4330 are consistent with NUREG/CR-3539 and other similar containment leakage risk studies:

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

EPRI TR-105189 (Reference 14)

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

NUREG-1493 (Reference 6)

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

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.

<u>EPRI TR-104285 (Reference 2)</u>

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..."

<u>NUREG-1150 (Reference 15) and NUREG/CR-4551 (Reference 7)</u>

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 Peach Bottom. The ex-plant calculation can be delineated to total person-rem for each identified Accident Progression Bin (APB) from NUREG/CR-4551. It should be noted that in the analysis for the Fermi 2 ILRT

extension, offsite dose consequence information from the SAMA analysis conducted for Fermi 2 License Renewal Application (References 27 and 28) is utilized in this analysis.

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 Fermi 2 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.

4.1 Plant Specific Inputs

The plant-specific information used to perform the Fermi 2 ILRT Extension Risk Assessment includes the following:

- Level 1 model results (Reference 17)
- Level 2 model results (Reference 17)

- Offsite Population Dose estimates from the SAMA Analysis (Reference 27)
- Release category definitions used in the Level 2 Model (Reference 17)
- ILRT results to demonstrate adequacy of the administrative and hardware issues (Reference 39). [1]
- Containment failure probability data (Reference 17)
- Information regarding the technical adequacy of the Fermi 2 model to address this application (see Appendix A)

4.1.1 Level 1 Model

The Level 1 PSA model that is used for Fermi 2 is characteristic of the as-built plant. The current Level 1 model is a linked fault tree model, and was quantified with the total Core Damage Frequency (CDF) of 1.73E-6/yr.

When each Level 2 release term was quantified, the results of the accident sequence sub-class IIID (LOCA with failure of vapor suppression capability) revealed a release term that was slightly greater (by 1.2E-9/yr) than the CDF for that accident sequence sub-class; this difference is attributable to success term conservatisms in the Level 2 model and the application of the min-cut upper bound (MCUB) approximation when calculating the combined CDF value from the sum of the individual cutsets. This difference in between the CDF and release term solution for sub-class IIID was added to the quantified CDF value and a result of 1.74E-6/yr was obtained; therefore, a value of 1.74E-6/yr is conservatively utilized for the ILRT risk calculations (it should be noted that this increase is less than 1% of the quantified CDF value).

Table 4-1 provides a summary of the internal events (including internal flooding) initiating event contribution CDF. Table 4-2 provides a summary of the internal events (including internal flooding) accident classification contribution to CDFF.

Table 4-1 – Initiating Events CDF Contribution

Initiator Group	Total IE Group Probability	%CDF
General Transients (without LOSP)	5.19E-07	29.9%
LOCAs	3.94E-07	22.7%
Internal Flood	3.06E-07	17.7%
LOSP	2.25E-07	13.0%
Special Initiators	1.59E-07	9.2%
Partial LOSP	1.31E-07	7.6%

[1] The two most recent Type A tests at Fermi 2 have been successful, so Type A test interval requirement is 10 years.

Table 4-2 – Accident Class CDF Contribution

Class	Accident Class	Frequency	%CDF
Class I	Loss of Makeup	9.83E-07	56.6%
Class II	Loss of DHR	1.64E-07	9.4%
Class III	LOCA	3.12E-07	17.9%
Class IV	ATWS	2.18E-07	12.6%
Class V	Containment Bypass	5.97E-08	3.4%

The Fermi 2 PRA model of record addresses internal events (including internal flooding) only. Internal fire, seismic, and other external hazards are addressed in Section 5.4.2.

4.1.2 Level 2 Model

The Level 2 Model that is used for Fermi 2 was developed to calculate the LERF contribution as well as the other release categories evaluated in the model. Table 4-3 defines the release terms by magnitude and timing. Table 4-4 summarizes the pertinent Fermi 2 results in terms of release category.

The quantified value for LERF (H/E) release at a 1.0E-12/yr truncation limit is 5.07E-12/yr. A comparison of the Level 1 and Level 2 accident class contribution results indicated that a small adjustment was necessary to LERF to ensure that releases for accident classes where an "intact" release was not deemed to be credible (i.e. Class II, Class IV, and Class V); the difference in the core damage contributions and the total release for these accident classes was incorporated into the LERF release term; as a result the LERF for the calculation of release terms in the ILRT release calculations is 5.25E-7/yr (an approximately 3% increase over 5.07E-7/yr).

Table 4-3 – Release Severity and Timing Classification

Release Severity Term Release Fraction		Release Timing	
Classification Category	Cs Iodide % in Release	Classification Category	Time of Release(1)
High (H)	Greater than 10	Late (L)	Greater than 24 hours
Moderate (M)	1 to 10	Intermediate (I)	4 to 24 hours
Low (L)	0.1 to 1	Early (E)	Less than 4 hours
Low-low (LL)	Less than 0.1		
Intact (OK)	<< 0.1		

Table 4-4 – Level 2 PSA Model Release Categories and Frequencies

Release Category	Description	Frequency (yr^{-1})
LERF	Large Early Release	5.25E-07
H/I	High Intermediate Release	9.83E-08
H/L	High Late Release	2.21E-10
M/E	Medium Early Release	2.12E-08
M/I	Medium Intermediate Release	1.21E-08
M/L	Medium Late Release	0.00E+00
L/E	Low Early Release	7.56E-08
L/I	Low Intermediate Release	4.62E-08
L/L	Low Late Release	0.00E+00
LL/E	Low-Low Early Release	3.15E-10
LL/I	Low-Low Intermediate Release	4.92E-08
LL/L	Low-Low Late Release	6.43E-11
CI	Containment Intact	9.09E-07
Release Total	Total Level 2 Release (Not Intact)	8.28E-07
CDF	Core Damage Frequency	1.74E-06

4.1.3 Population Dose Calculations

The population dose values (in terms of person-rem) reported in the Severe Accident Management Alternative (SAMA) Analysis and the License Renewal Application (References 27 and 28) were utilized for the ILRT application; this information is presented in Table 4-5.

The frequencies for each release category in Table 4-4 are then used to calculate a "population dose risk" term, which is the product other the release frequency from Table 4-4 and the population dose. The results of this calculation are presented in Table 4-5.

It should be noted that the LERF (H/E) frequency term was decomposed for the purpose of dose rate calculations into BOC (break outside containment) and non-BOC terms. This was done by subtracting the LERF contribution from Class V release term as well as the Class 2 large containment isolation failure term calculated from the methodology obtained from Section 4.3 of Reference 26. The Class 2 containment isolation failure frequency ($F_{Class\ 2}$) is

$$F_{Class\ 2} = PROB_{large\ CI} * CDF_{Total} \qquad \text{(Eqn. 4-1)}$$

Where,

$PROB_{large\ CI}$ = random containment large isolation failure probability (large valves); the parameter value (8.60E-4) is taken to equal the Fermi 2 Bayesian-updated probability for a safety-related motor-operated valve failing to close on demand found in Reference 18 (Fermi 2 Component Data Notebook, Vol. 1, Revision 1).

CDF_{Total} = total plant-specific core damage frequency.

The LERF (H/E) release term associated with break outside containment sequences ($LERF_{BOC}$) and non-BOC sequences ($LERF_{NON\text{-}BOC}$) are calculated as follows:

$$LERF_{BOC} = LERF_{Class\ V} + F_{Class\ 2} \qquad \text{(Eqn. 4-2)}$$

$$LERF_{NON\text{-}BOC} = LERF_{Total} - LERF_{BOC} \qquad \text{(Eqn. 4-3)}$$

Where,

$LERF_{Total}$ = total plant-specific large early release frequency.

$LERF_{Class\ V}$ = large early release frequency associated with Class V (containment bypass)

Numerically,

$F_{Class\ 2}$ = 8.60E-4 * 1.74E-6/yr = 1.49E-9/yr

$LERF_{BOC}$ = 5.97E-8/yr + 1.49E-9/yr = 6.12E-8/yr

$LERF_{NON\text{-}BOC}$ = 5.25E-7/yr - 6.12E-8/yr = 4.64E-7/yr

Table 4-5 – Population Dose and Dose Risk Calculations

Release Category	Population Dose (Person-Rem)	Frequency (yr^{-1})	Population Dose Risk (Person-Rem / yr)
$LERF_{BOC}$	2.18E+07	6.12E-08	1.33E+00
$LERF_{NON\text{-}BOC}$	8.10E+06	4.64E-07	3.76E+00
H/I	9.52E+06	9.83E-08	9.36E-01
H/L	8.98E+06	2.21E-10	1.98E-03
M/E	2.48E+06	2.12E-08	5.26E-02
M/I	2.76E+06	1.21E-08	3.34E-02
L/E	2.26E+05	7.56E-08	1.71E-02
L/I	2.14E+06	4.62E-08	9.89E-02
LL/E	1.31E+04	3.15E-10	4.13E-06
LL/I	1.29E+05	4.92E-08	6.35E-03
LL/L	1.29E+05	6.43E-11	8.29E-06
CI	6.46E+01	9.09E-07	5.87E-05

4.1.4 Release Category Definitions

Table 4-6 defines the accident classes used in the ILRT extension evaluation, which is consistent with the EPRI methodology (Reference 26). These containment failure classifications are used in Section 5 of this analysis to determine the risk impact of extending the Containment Type A test interval.

Table 4-6 – EPRI Containment Failure Classification

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 La, under Appendix J for that plant.
2	Containment isolation failures (as reported in the IPEs) include 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 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 4 isolation failures, but is applicable to sequences involving Type C tests and their potential failures.
6	Containment isolation failures include 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.

4.1.5 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 4-6, is divided into two sub-classes, Class 3a and Class 3b, representing small and large leakage failures respectively.

The probability of the EPRI Class 3a and Class 3b failures is determined consistent with the EPRI Guidance (Reference 26). 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 ΔLERF 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 Fermi 2 is detailed in Section 5.0.

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.

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

An estimate of the likelihood and risk impact 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.

Section 5.4.1 fully discusses the methodology and the results as it pertains to the Fermi 2 ILRT extension.

5. Detailed Calculation

The application of the approach based on the guidance contained in EPRI Report No. 1009325, Revision 2-A, Appendix H (Reference 26), EPRI TR-104285 (Reference 2) and previous risk assessment submittals on this subject (References 5, 8, 21, 22, and 23) have led to the results as described in this section. The results are displayed according to the eight accident classes defined in the EPRI report, as described in Table 5-1.

The analysis performed examined Fermi 2-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).
- Accident sequences involving containment bypassed (EPRI TR-104285, Class 8 sequences), large containment isolation failures (EPRI TR-104285, Class 2 sequences), and small containment isolation "failure-to-seal" events (EPRI TR-104285, Class 4 and 5 sequences) 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.

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 5-1 - Develop plant-specific person-rem dose (population dose) per reactor year for each of the eight accident classes.

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) and the impact on the Conditional Containment Failure Probability (CCFP).

Step 5 – Perform sensitivity analysis.

Table 5-1 – Accident Classes (Containment Release Types)

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)

5.0 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). 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 5-1 were developed for Fermi 2 by first determining the frequencies for Classes 1, 2, 7 and 8 using the categorized sequences and the identified correlations shown in Table 5-3, scaling these frequencies to account for the uncategorized sequences, determining the frequencies for Classes 3a and 3b, and then determining the remaining frequency for Class 1. It should be noted that per the rationale presented in EPRI TR-104285, EPRI Classes 4, 5, and 6 are not applicable to the quantification of the risk associated with an increase in ILRT Type A testing frequency.

The development of the equations used and the method by which each EPRI class is addressed is contained Section 4.3 of Reference 26. Below are several explanatory notes outlining interpretations which are not specifically stated in that reference or in Table 5-3.

1. Class 1: Represented in Table 5-3 by (CDF_{total} - Release $Frequency_{total}$ – $F_{Class\ 3a}$ – $F_{Class\ 3b}$), where

    a) CDF_{total} represents the total core damage frequency calculated from the Fermi 2 model of record (Reference 17) at a truncation limit of 1.0E-12/yr; the value is obtained from the final row in Table 5-2. [2]
    b) Release $Frequency_{total}$ is the Level 2 model release frequency (both LERF and non-LERF sequences truncated at 1.0e-12/yr) as described in the Fermi 2 PRA model of record (Reference 17) and associated documentation; the value is taken from the final row in Table 5-2.
    c) $F_{Class\ 3a}$ is the frequency for Class 3a releases as defined in Item 3b below.
    d) $F_{Class\ 3b}$ is the frequency for Class 3b releases as indicated in Item 4b below.

2. Class 2: This group consists of all core damage accident progression bins for which a failure to isolate the containment occurs following the onset of core damage (as opposed to Class 8, where the containment is bypassed as part of the CDF accident sequence). The Class 2 frequency is estimated by multiplying total CDF by the probability of containment isolation failure (represented by the plant PRA-specific probability of a motor-operated valve to fail to close).

3. Class 3a:
    a) From Reference 26 Section 5.2.1, the class 3a leakage probability ($PROB_{Class\ 3a}$) is based on data from the ILRT testing data (Section 3.5), which is two "small" failures in 217 tests (2/217=0.0092).
    b) The formulation of the Class 3a frequency ($F_{Class\ 3a}$) involves the more detailed treatment presented in Section 5.2.1 of Reference 26:

    $$F_{Class\ 3a} = (CDF_{total} - CDF_{AONL}) * PROB_{Class\ 3a} \qquad \text{(Eqn. 5-1)}$$

    Where,

    CDF_{AONL} = Always or Never LERF CDF; this parameter is quantified by the formula:

[2] Table 5-2 is a summary of the Level 2 model release terms as sorted by core damage accident classes/sub-classes. CDF, LERF, and Total Level 2 Release terms are displayed in this table.

$$CDF_{AONL} = CDF_{Class\ V} + CDF_{Class\ IV} + CDF_{Class\ IBL} + CDF_{Class\ IIID} \quad \text{(Eqn. 5-2)}$$

Where, $CDF_{Class\ V}$, $CDF_{Class\ IV}$, $CDF_{Class\ IBL}$, and $CDF_{Class\ IIID}$ are the release terms as defined in Table 5-2 (where it is demonstrated that the conditional large early release probability is either 1.0 or 0.0)

Numerically,

CDF_{AONL} = 5.97E-08/yr + 2.18E-07/yr + 1.58E-08 + 6.66E-08 = 3.60E-07/yr

$F_{Class\ 3a}$ = (1.74E-06/yr - 3.60E-07/yr) * 0.0092 = 1.27E-08/yr

4. Class 3b:
   a) From Reference 26 Section 5.2.1, the class 3b failure probability is based on the Jeffrey's Non-Informative Prior and is equal to 0.0023.
   b) The formulation of the Class 3b frequency ($F_{Class\ 3b}$) involves the more detailed treatment presented in Section 5.2.1 of Reference 26; this treatment is the same as is outlined above for Class 3a.

$$F_{Class\ 3b} = (CDF_{total} - CDF_{AONL}) * PROB_{Class\ 3b} \quad \text{(Eqn. 5-3)}$$

Numerically,

$F_{Class\ 3b}$ = (1.74E-06/yr - 3.60E-07/yr) * 0.0023 = 3.16E-09/yr

5. Class 7: This class encompasses phenomenologically-induced containment failures (both early and late in the sequence). This is interpreted to encompass all of the accident classes not assigned to other EPRI categories (those not involving containment bypass and containment isolation failure). The frequency for this class is the sum of all the release frequencies in Table 4-5 with the exception of $LERF_{BOC}$.

6. Class 8: This group consists of all core damage accident progression bins in which containment bypass occurs. Each plant's PRA is used to determine the containment bypass contribution. Contributors to bypass events include ISLOCA events and break outside containment (BOC) events with an isolation failure. Class 8 in the Fermi 2 containment model is represented by the CDF Accident Class V (containment bypass).

Table 5-2 – Core Damage, LERF, and Total Release by Accident Class

<table>
<tr><th>PDS Class</th><th>PDS Description</th><th>PDS Subclass</th><th>Subclass CDF [yr-1]</th><th>Class CDF [yr-1]</th><th>LERF [yr-1]</th><th>Total Release Frequency [yr-1]</th></tr>
<tr><td rowspan="5">I
(Loss of Makeup)</td><td>Loss of inventory makeup in which the reactor pressure remains high</td><td>A</td><td>3.94E-07</td><td rowspan="5">9.83E-07</td><td>4.84E-08</td><td>9.76E-08</td></tr>
<tr><td rowspan="2">SBO and loss of inventory makeup (BE: CD at < 4 hrs, BL: CD > 4 hrs)</td><td>IBE</td><td>8.02E-09</td><td>1.42E-09</td><td>4.33E-09</td></tr>
<tr><td>IBL</td><td>6.66E-08</td><td>0.00E+00</td><td>5.84E-08</td></tr>
<tr><td>Loss of coolant inventory induced by ATWS with containment intact</td><td>C</td><td>3.56E-07</td><td>1.42E-08</td><td>5.79E-08</td></tr>
<tr><td>Loss of coolant inventory makeup in which reactor pressure reduced to 200 psi</td><td>D</td><td>1.58E-07</td><td>9.39E-09</td><td>9.45E-08</td></tr>
<tr><td>II
(Loss of Decay Heat Removal)</td><td>Loss of containment heat removal causes core damage (containment either failed or vented)</td><td>II</td><td>1.64E-07</td><td>1.64E-07</td><td>1.43E-07</td><td>1.64E-07</td></tr>
<tr><td rowspan="4">III
(LOCA)</td><td>CD initiated by vessel rupture where containment integrity is not breached in the initial time phase of the accident</td><td>A</td><td>0.00E+00</td><td rowspan="4">3.12E-07</td><td>0.00E+00</td><td>0.00E+00</td></tr>
<tr><td>Small/Medium LOCAs for which the reactor cannot be depressurized prior to CD</td><td>B</td><td>2.25E-07</td><td>1.29E-08</td><td>4.13E-08</td></tr>
<tr><td>Medium/Large LOCAs for which the reactor is at low pressure and no effective injection is available</td><td>C</td><td>7.14E-08</td><td>1.72E-09</td><td>1.65E-08</td></tr>
<tr><td>LOCA/RPV failure, vapor suppression is inadequate challenging containment integrity with subsequent failure of makeup</td><td>D</td><td>1.58E-08</td><td>1.58E-08</td><td>1.58E-08</td></tr>
</table>

PDS Class	PDS Description	PDS Subclass	Subclass CDF [yr^{-1}]	Class CDF [yr^{-1}]	LERF [yr^{-1}]	Total Release Frequency [yr^{-1}]
IV (ATWS)	Failure of adequate shutdown reactivity with RPV initially intact; CD induced post containment failure	A	2.18E-07	2.18E-07	2.18E-07	2.18E-07
	Failure of adequate shutdown reactivity with RPV initially breached; CD induced post containment failure	L	0.00E+00		0.00E+00	
	Failure of adequate shutdown reactivity with RPV initially intact; CD induced post high containment pressure (Not used)	T	0.00E+00		0.00E+00	
	Class IVA or IVL except that the vent operates as designed; loss of makeup occurs sometime following vent initiation. Suppression pool saturated but intact (Not used)	V	0.00E+00		0.00E+00	
V (Containment Bypass)	Unisolated LOCA outside containment	N/A	5.97E-08	5.97E-08	5.97E-08	5.97E-08
All	**Total**		1.74E-06	1.74E-06	5.25E-07	8.28E-07

Table 5-3 – EPRI Containment Release Type Tabulation

EPRI Class	Description	Formula (Resultant Frequency) [3]	Resultant Frequency $[yr^{-1}]$ [4]	Dose [person-rem]	Dose Rate [person-rem/yr] [5]
1	Containment Intact	CDF_{total}-Release $Frequency_{total}$ - $F_{class\ 3a}$ -$F_{class\ 3b}$	8.92E-07 [2]	64.6 [6]	5.76E-05 [2]
2	Large Containment Isolation Failures [1]	$PROB_{large\ CI}$ * CDF_{total}	1.49E-09	2.18E+07 [7]	3.25E-02
3a	Small Pre-Existing Leak in Containment	$PROB_{Class\ 3a}$ * (CDF_{total} - CDF_{AONL})	1.27E-08	646 [8]	8.18E-06
3b	Large Pre-Existing Leak in Containment	$PROB_{Class\ 3b}$*(CDF_{total} - CDF_{AONL})	3.16E-09	6460 [9]	2.04E-05
7	Severe accident phenomena-induced containment failures (early and late)	Σ(All release categories except intact and BOC from Table 4-5) [11]	7.67E-07	6.39E+06 [10]	4.90E+00 [11]
8	Containment Bypass sequences	$CDF_{Class\ V}$	5.97E-08	2.18E+07 [7]	1.30E+00

Notes:

(1) Random Containment Large Isolation failure probability, $PROB_{large\ CI}$, is taken to equal the Fermi 2 Bayesian-updated probability for a safety-related motor-operated valve failing to close on demand (Fermi 2 Component Data Notebook, Vol. 1, Revision 1), 8.60E-4.

(2) EPRI Class 1 "containment intact" term in this table differs slightly from the value listed in Table 4-5 due to the subtraction of the frequencies for Class 3a and Class 3b in this table.

(3) Formulas based upon information contained in Section 5.0.

(4) Frequencies calculations are based upon equations in the "Formula (Resultant Frequency)' column and the written description for each accident class contained in Section 5.0.

(5) Unless otherwise noted, dose rates are calculated by multiplying the values in the "Resultant Frequency" and "Dose" columns.

(6) Value obtained from the "CI" release term in Table 4-5.

(7) Value obtained from the "LERF-BOC" release term in Table 4-5.

(8) Pre-existing small leak population dose equal to 10 times EPRI accident Class 1 population dose per Section 5.2.2 of Reference 26.

(9) Pre-existing large leak population dose equal to 100 times EPRI accident Class 1 population dose per Section 5.2.2 of Reference 26.

(10) Dose for Class 7 is obtained by dividing the "Dose Rate" result by the "Resultant Frequency" result.

(11) Dose for Class 7 is obtained by summing the product of the "Frequency" and "Population Dose Risk" columns for each release category (with the exception of LERF-BOC and CI) in Table 4-5.

5.1 Step 2 – Develop Plant-Specific Person-Rem Dose (Population Dose) Per Reactor Year

Plant-specific release analyses were performed to estimate the person-rem doses to the population within a 50-mile radius from the plant. These releases are presented and discussed in Section 4.1.3 and Table 4-5. The population dose values reported in the Severe Accident Management Alternative (SAMA) Analysis in the License Renewal Application (References 27 and 28).

5.2 Step 3 - Evaluate Risk Impact of Extending Type A Test Interval to once per 15 years

In this step, the risk impact associated with the change in ILRT testing intervals is evaluated in terms of changes to the accident class frequencies and populations doses. This is accomplished in a three-step process (these steps are referred to as sub-steps to distinguish them from the overall steps).

In the first sub-step, the change in probability of leakage detectable only by ILRT (Classes 3a and 3b) for the new surveillance intervals of interest is determined. NUREG 1493 (Reference 5) states that relaxing the ILRT frequency from three in 10 years to one in 10 years will increase the average time that a leak that is detectable only by ILRT goes undetected from 18 to 60 months (1/2 the surveillance interval), a factor of 60/18 = 3.33 increase. Therefore, relaxing the ILRT testing frequency from three in 10 years to one in 15 years will increase the average time that a leak that is detectable only by ILRT goes undetected from 18 to 90 months (1/2 the surveillance interval), a factor of 90/18 = 5.0 increase.

In the second sub-step, the population dose rate for the new surveillance intervals of interest is determined by multiplying the dose by the frequency for each of the accident classes. The accident class dose rates are then summed to obtain the total dose rate.

The results of the aforementioned sub-steps are contained in Table 5-4.

Table 5-4 - Accident Class Frequency and Population Doses versus ILRT Frequency

EPRI Accident Class	ILRT Frequency					
	3 per 10 years		1 per 10 years		1 per 15 years	
	Frequency [yr^{-1}]	Person-rem/yr	Frequency [yr^{-1}]	Person-rem/yr	Frequency [yr^{-1}]	Person-rem/yr
1	8.92E-07	5.76E-05	8.55E-07	5.53E-05	8.29E-07	5.35E-05
2	1.49E-09	3.25E-02	1.49E-09	3.25E-02	1.49E-09	3.25E-02
3a	1.27E-08	8.18E-06	4.22E-08	2.73E-05	6.33E-08	4.09E-05
3b	3.16E-09	2.04E-05	1.05E-08	6.81E-05	1.58E-08	1.02E-04
7	7.67E-07	4.90E+00	7.67E-07	4.90E+00	7.67E-07	4.90E+00
8	5.97E-08	1.30E+00	5.97E-08	1.30E+00	5.97E-08	1.30E+00
Totals	1.74E-06	6.24E+00	1.74E-06	6.24E+00	1.74E-06	6.24E+00

The effects of postulated containment liner corrosion are contained in Section 5.4.1.

5.3 Step 4 – Evaluate Change in LERF and CCFP

In this step, the changes in LERF and CCFP as a result of the evaluation of extended ILRT intervals are evaluated.

The risk associated with extending the ILRT interval involves a potential that a core damage event that normally would result in only a small radioactive release from containment could result in a large release due to an undetected leak path existing during the extended interval. As discussed in References 1 and 2, only Class 3 sequences have the potential to result in early releases if a pre-existing leak were present. Late releases are excluded regardless of size of the leak because late releases are not, by definition, LERF events. The frequency of class 3b sequences is used as a measure of LERF, and the change in LERF is determined by the change in class 3b frequency. Refer to Regulatory Guide 1.174 (Reference 4) for LERF acceptance guidelines. Delta LERF is determined using the equation below, where the “frequency of class 3b new interval x” is the frequency of the EPRI accident class 3b for the ILRT interval of interest and the “frequency of class 3b baseline” is defined as the EPRI accident class 3b frequency for ILRTs performed on a three-per-10-years basis.

ΔLERF = (freq. of class 3b new interval x) - (freq. of class 3b baseline) (Eqn. 5-4)

The conditional containment failure probability (CCFP) is defined as the probability of containment failure given the occurrence of a core damage accident, which is expressed as:

$$CCFP = [1 - (F_{Class\ 1} + F_{class\ 3a}) / CDF] * 100\% \qquad \text{(Eqn. 5-5)}$$

Where,

$F_{Class\ 1}$ is the frequency of EPRI Class 1 release from the “frequency” columns in Table 5-4.

$F_{Class\ 3a}$ is the frequency of EPRI Class 3a release from the "frequency" columns in Table 5-4.

The increase in CCFP is expressed as:

ΔCCFP = (CCFP at ILRT revised interval) – (CCFP at baseline interval) (Eqn. 5-6)

Table 5-5 – Fermi 2 Delta LERF and CCFP

Risk Metric	ILRT Testing Frequency		
	3 in 10 years	1 in 10 years	1 in 15 years
ΔLERF	N/A	7.37E-09	1.27E-08
CCFP	47.88%	48.30%	48.61%
ΔCCFP	N/A	0.42%	0.73%

5.4 Step 5 – Evaluate Sensitivity of Results

In this step, the risk impact results sensitivity to assumptions in liner corrosion, the impact of external events, and containment overpressure credit are investigated.

In evaluating the impact of liner corrosion on the extension of ILRT testing intervals, the Calvert Cliffs methodology is used. The methodology developed for Calvert Cliffs investigates how an age-related degradation mechanism can be factored into the risk impact associated with longer ILRT testing intervals.

An assessment of the impact of external events is performed. The primary basis for this investigation is the determination of the total LERF following an increase in the ILRT testing frequency from three times in 10 years to once per 15 years.

Although containment overpressure is not credited in the design basis for Fermi 2, it is credited in a subset of PRA scenarios. Therefore, sensitivity studies will be performed on containment overpressure.

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

This sensitivity study presents an estimate of the likelihood and risk implications of corrosion-induced leakage of steel containment liners being undetected during the extended ILRT test intervals evaluated in this report. The methodology employed in this sensitivity case is taken from the Calvert Cliffs liner corrosion analysis (References 5 and 20). The Calvert Cliffs analysis is performed for a concrete cylinder and dome with a concrete basemat, each with a steel liner. The Fermi 2 containment is a BWR Mark I type with a steel shell in the drywell and wetwell (torus) regions. The shell in the drywell is surrounded by a concrete shield.

The following approach is used to determine the change in likelihood, due to extending the ILRT interval, of detecting corrosion of the steel liner. This likelihood is used to determine the potential change in risk in the form of a sensitivity case. Consistent with the Calvert Cliffs analysis, the following are addressed:

1. Differences between the containment basemat and other regions of the containment
2. The historical steel liner/shell flaw likelihood due to concealed corrosion
3. The impact of aging
4. The likelihood that visual inspections will be effective in detecting a flaw

The assumptions used in this sensitivity study are consistent with the Calvert Cliffs methodology and include the following:

1. A half failure is assumed for the basemat concealed liner corrosion due to lack of identified failures.
2. Two corrosion events are used to estimate the liner flaw probability. These events, one at North Anna Unit 2 and the other at Brunswick Unit 2, were initiated from the non-visible (backside) portion of the containment liner.
3. Consistent with the Calver Cliff's analysis, the estimated historical flaw probability is limited to 5.5 years to reflect the years since September 1996, when 10CFR50.55a started requiring visual inspections. Additional success data were not used to limit the aging impact of the corrosion issue, even though inspections were being performed prior to this data (and have been performed since the timeframe of the Calvert Cliffs analysis) and there has been no evidence that additional corrosion issues were identified.
4. Consistent with the Calvert Cliffs analysis, the corrosion-induced steel liner/shell flaw likelihood is assumed to double every five years. This is based solely on judgment and is included in this analysis to address the increase in likelihood of corrosion as the steel shell ages.
5. 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 that corresponds to the ILRT pressure of 37 psig. For Fermi 2, the containment failure probabilities are conservatively assumed to be 10% for the shell wall and 1% for the basemat.
6. In the Calvert Cliffs analysis, it is noted that approximately 85% of the interior wall surface is accessible for visual inspections. At Fermi 2, it is estimated that the area accessible for visual inspections is 96% per Reference 38; note that the uninspectable area is under the drywell floor. Given that the flaw is visible, a 5% visual inspection detection failure likelihood and a total detection failure likelihood of 10% are used. To date, all liner corrosion events have been detected through visual inspection.
7. All non-detectable failures are assumed to result in early releases. This approach is conservative and avoids detailed analysis of containment failure timing and operator recovery actions.

The results of the liner corrosion analysis for the purpose of the calculation of the likelihood of non-detected containment leakage described above are summarized in Table 5-6.

Table 5-6 – Liner Corrosion Analysis

Step	Description	Containment (Excluding Basemat)	Containment Basemat
1	Historical Steel Liner Flaw Likelihood / Failure Data	Events: 2 2 / (70 * 5.5) 5.19E-03	Events: 0 (assume half a failure) 0.5 / (70 * 5.5) 1.30E-03
	Failure Data (1)	5.19E-03	1.30E-03
2	Age-Adjusted Steel Liner Flaw Likelihood (Year 1) (2)	2.05E-03	5.13E-04
	Age-Adjusted Steel Liner Flaw Likelihood (Avg Years 5-10) (2)	5.20E-03	1.30E-03
	Age-Adjusted Steel Liner Flaw Likelihood (Year 15) (2)	1.43E-02	3.59E-03
	Age-Adjusted Steel Liner Flaw Likelihood (Avg 15 Years) (2)	6.45E-03	1.61E-03
3	Flaw Likelihood (3 Years) (3a)	0.71%	0.18%
	Flaw Likelihood (10 Years) (3a)	4.14%	1.04%
	Flaw Likelihood (15 Years) (3a)	9.68% (3b)	2.42% (3c)
4	Likelihood of Breach in Containment Given Steel Liner Flaw (4)	1.00%	0.10%
5	Visual Inspection Detection Failure Likelihood	10.00% (5a)	100.00% (5b)
6	Likelihood of Non-Detected Containment Leakage (3 yr)	7.12E-06	1.78E-06
	Likelihood of Non-Detected Containment Leakage (10 yr)	4.14E-05	1.04E-05
	Likelihood of Non-Detected Containment Leakage (15 yr)	9.68E-05	2.42E-05

Notes:

(1) Containment location specific (consistent with Calvert Cliffs analysis).

(2) During 15-year interval, assume failure rate doubles every five years (14.9% increase per year). The average for fifth to tenth year set to the historical failure rate (consistent with Calvert Cliffs analysis).

(3) (a) Uses age-adjusted liner flaw likelihood (Step 2), assuming failure rate doubles every five years (consistent with Calvert Cliffs).

(b) Note that the Calvert Cliffs analysis presents the delta between three and 15 years of 8.7% to utilize in the estimation of the delta-LERF value. For this analysis, however, the values are

calculated based on three-, 10-, and 15-year intervals, consistent with the desired presentation of the results.

(c) Note that the Calvert Cliffs analysis presents the delta between three and 15 years of 2.2% to utilize in the estimation of the delta-LERF value. For this analysis, however, the values are calculated based on the three-, 10-, and 15-year intervals, consistent with the desired presentation of the results.

(4) The failure probability of the cylinder and dome is assumed to be 1%, and basemat is 0.1% as compared to 1.1% and 0.11% in the Calvert Cliffs analysis.

(5) (a) 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. Five percent visible failure detection is a conservative assumption.

(b) Cannot be visually inspected.

The cumulative likelihood of non-detected containment leak ($P_{leak\text{-}corr\text{-}tot}$) due to corrosion is the sum in Step 6 for the containment walls and the containment basemat:

1. At 3 years: 7.12E-06 + 1.78E-06 = 8.89E-06
2. At 10 years: 4.14E-05 + 1.04E-05 = 5.18E-05
3. At 15 years: 9.68E-05 + 2.42E-05 = 1.21E-04

The above factors are applied to those core damage accidents that are not already independently LERF or that could never result in LERF. For example, the three-in-10-years case is calculated as follows:

1. Per Table 5-4, the EPRI Class 3b frequency is 3.16E-09/yr.
2. As discussed in Section 5.2.1, the Fermi 2 CDF associated with accidents that are guaranteed to be LERF or could never result in LERF (CDF_{AONL}) per Eqn. 5-2 is 3.60E-07/yr. Thus, the value for "CDF_{total} - CDF_{AONL}" per Eqn. 5-3 is 1.74E-06/yr – 3.60E-07/yr = 1.38E-06/yr
3. The increase in the base case Class 3b frequency ($\Delta F_{3b\text{-}corr}$) due to the corrosion-induced concealed flaw issue is calculated based upon the formula:

$$\Delta F_{3b\text{-}corr} = (CDF_{total} - CDF_{AONL}) * P_{leak\text{-}corr\text{-}tot} \qquad \text{(Eqn. 5-7)}$$

   Numerically, this corresponds to 1.38E-06*8.89E-06=1.22E-11/yr, where 8.89E-6 was previously shown to be the cumulative likelihood of non-detected containment leakage due to corrosion at three years.
4. The three-in-10-years Class 3b frequency, including the corrosion-induced concealed flaw issue, is calculated as the sum of the baseline class 3b frequency without corrosion ($F_{Class\ 3b}$) from Eqn. 5-3 and the increase in class 3b frequency ($\Delta F_{3b\text{-}corr}$) due to corrosion from Eqn. 5-7 (3.16E-09/yr+1.22E-11/yr=3.18E-09/yr).

Table 5-7, Table 5-8, and Table 5-9 provide a summary of the base case as well as the corrosion sensitivity case for various ILRT intervals (three times per 10 years, once per 10 years, and once per 15 years, respectively).

Each of tables is sub-divided further into corrosion and non-corrosion cases. For both the corrosion and non-corrosion cases, the frequencies of the EPRI accident classes are

provided. In the corrosion cases, an additional column titled "Delta person-rem per yr" is provided. The "Δ Person-rem / yr" column provides the change in person-rem per year between the corrosion and non-corrosion cases. Negative values in the "Δ Person-rem / yr" column indicate a reduction in the person-rem per year for the selected accident class; this occurs only in EPRI accident Class 1 and is a result of the reduction in the frequency of the accident Class 1 and an increase in accident Class 3b. Rows for the totals, both frequency and dose rate, are provided in the tables. Additional summary rows are also provided.

1. The change in dose rate, expressed as person-rem/yr and percentage of the total base dose is provided in the row below the "total" row.
2. The Conditional Containment Failure Probability (CCFP) is provided in the next row, followed by the absolute change in CCPF in percentage points.
3. Class 3b LERF is also provided and indicates the accident class 3b frequency as well as the change in the class 3b frequency in parentheses. This difference is calculated between the non-corrosion and corrosion cases.
4. The next row titled "Delta LERF from Base Case (3 per 10 years)", which is not contained in Table 5-7, provides the change in LERF as a function of ILRT frequency from the base case. The difference between the non-corrosion and corrosion cases is contained in the last column underneath the heading "Difference".
5. The last row of the table titled "Delta LERF From 1 per 10 Years", which is only contained in Table 5-9 provides the change in LERF as a result of changing the ILRT frequency from one in 10 years to one in 15 years. The difference between the non-corrosion and corrosion cases is contained in the last column underneath the heading "Difference".

The principal sensitivity analysis in this section presents an estimate of the likelihood and risk implications of corrosion-induced leakage of steel containment liners not being detected during the extended ILRT test intervals evaluated in this report. The analysis considers ILRT extension time, inspections, and concealed degradation in uninspectable areas. As can be seen from Table 5-7, Table 5-8, and Table 5-9, the change from the base case of three tests per 10 years to one test per 15 years in LERF with corrosion is very small (1.3E-08/yr). Similarly, the change in delta-LERF between the corrosion and non-corrosion cases for one per 15 years is, correspondingly, very small at 1.5E-10/yr.

Table 5-7 – Summary of Fermi 2 Corrosion Analysis (3 per 10 yr ILRT Frequency)

EPRI Accident Class	ILRT Frequency				
	3 per 10 years				
	Without Corrosion		With Corrosion		
	Frequency $[yr^{-1}]$ (1)	**Person-rem/yr (1)**	**Frequency $[yr^{-1}]$ (3)**	**Person-rem/yr (3)**	**Δ Person-rem / yr (8)**
1	8.92E-07	5.76E-05	8.92E-07 (6)	5.76E-05 (7)	-7.91E-10
2	1.49E-09	3.25E-02	1.49E-09	3.25E-02	0.00E+00
3a	1.27E-08	8.18E-06	1.27E-08	8.18E-06	0.00E+00
3b	3.16E-09	2.04E-05	3.18E-09 (4)	2.05E-05 (5)	7.91E-08
7	7.67E-07	4.90E+00	7.67E-07	4.90E+00	0.00E+00
8	5.97E-08	1.30E+00	5.97E-08	1.30E+00	0.00E+00
Total	**1.74E-06**	**6.24E+00**	**1.74E-06**	**6.24E+00**	**7.83E-08**
Δ Dose	N/A				
CCFP	4.79E-01 (2)		4.79E-01 (12)		
ΔCCFP	N/A				
Class 3b LERF	3.16E-09 (9)		3.18E-09 (10)		**Delta 3b LERF** 1.22E-11 (11)

Notes:

(1) Values taken from Table 5-4.
(2) Values taken from Table 5-5.
(3) Values for all classes except Class 1 and Class 3b are identical to the “without corrosion” values.
(4) Value calculated by summing “without corrosion” frequency and the ΔF value derived from Eqn. 5-7.
(5) Value obtained by multiplying the Class 3b “with corrosion” frequency calculated in this table and the dose for Class 3b listed in Table 5-3.
(6) Value calculated by using the formula for Class 1 listed in Table 5-3.
(7) Value obtained by multiplying the Class 1 “with corrosion” frequency calculated in this table and the dose for Class 1 listed in Table 5-3.
(8) The values in this column are the difference in dose rate (person-rem/yr) between the “with corrosion” and "without corrosion” cases.
(9) Value is identical to the Class 3b “without corrosion” frequency listed above.
(10) Value is identical to the Class 3b “with corrosion” frequency listed above.
(11) Value is the difference in the “with corrosion” and “without corrosion” Class 3b LERF values.
(12) Value calculated based upon Eqn. 5-5.

Table 5-8 – Summary of Fermi 2 Corrosion Analysis (1 per 10 yr ILRT Frequency)

EPRI Accident Class	ILRT Frequency				
	1 per 10 years				
	Without Corrosion		With Corrosion		
	Frequency [yr^{-1}] (1)	Person-rem/yr (1)	Frequency [yr^{-1}] (3)	Person-rem/yr (3)	Δ Person-rem / yr (8)
1	8.55E-07	5.53E-05	8.55E-07 (6)	5.52E-05 (7)	-4.60E-09
2	1.49E-09	3.25E-02	1.49E-09	3.25E-02	0.00E+00
3a	4.22E-08	2.73E-05	4.22E-08	2.73E-05	0.00E+00
3b	1.05E-08	6.81E-05	1.06E-08 (4)	6.85E-05 (5)	3.92E-07
7	7.67E-07	4.90E+00	7.67E-07	4.90E+00	0.00E+00
8	5.97E-08	1.30E+00	5.97E-08	1.30E+00	0.00E+00
Total	**1.74E-06**	**6.24E+00**	**1.74E-06**	**6.24E+00**	**3.88E-07**
Δ Dose	6.44E-05 (17)		6.47E-05 (18)		
CCFP	48.30% (2)		48.31% (12)		
ΔCCFP	0.42% (2)		0.43% (13)		
Class 3b LERF	1.05E-08 (9)		1.06E-08 (10)		**Difference** 7.13E-11 (11)
Delta Class 3b LERF (from 3 in 10 yr base case)	7.37E-09 (14)		7.43E-09 (15)		**Difference** 5.90E-11 (16)

Notes:

(1) Values taken from Table 5-4.
(2) Values taken from Table 5-5.
(3) Values for all classes except Class 1 and Class 3b are identical to the “without corrosion” values.
(4) Value calculated by summing “without corrosion” frequency and the ΔF value derived from Eqn. 5-7.
(5) Value obtained by multiplying the Class 3b “with corrosion” frequency calculated in this table and the dose for Class 3b listed in Table 5-3.
(6) Value calculated by using the formula for Class 1 listed in Table 5-3.
(7) Value obtained by multiplying the Class 1 “with corrosion” frequency calculated in this table and the dose for Class 1 listed in Table 5-3.
(8) The values in this column are the difference in dose rate (person-rem/yr) between the “with corrosion” and without corrosion” cases.
(9) Value is identical to the Class 3b “without corrosion” frequency listed above.
(10) Value is identical to the Class 3b “with corrosion” frequency listed above.
(11) Value is the difference in the “with corrosion” and “without corrosion” Class 3b LERF values.
(12) Value calculated based upon Eqn. 5-5.
(13) Value is the difference between the “with corrosion” CCFP in this table minus the “with corrosion” CCFP in Table 5-7.
(14) Value is the difference between the “without corrosion” Class 3b LERF in this table minus the “without corrosion” Class 3b LERF in Table 5-7.
(15) Value is the difference between the “with corrosion” Class 3b LERF in this table minus the “with corrosion” Class 3b LERF in Table 5-7.
(16) Value is the difference in the “with corrosion” and “without corrosion” Delta Class 3b LERF values.
(17) Value is the difference between the “without corrosion” Dose (person-rem/yr) in this table minus the “without corrosion” Dose in Table 5-7.
(18) Value is the difference between the “with corrosion” Dose (person-rem/yr) in this table minus the “with corrosion” Dose in Table 5-7.

Table 5-9 – Summary of Fermi 2 Corrosion Analysis (1 per 15 yr ILRT Frequency)

EPRI Accident Class	ILRT Frequency 1 per 15 years				
	Without Corrosion		With Corrosion		
	Frequency [yr^{-1}] (1)	Person-rem/yr (1)	Frequency [yr^{-1}] (3)	Person-rem/yr (3)	Δ Person-rem / yr (8)
1	8.29E-07	5.35E-05	8.29E-07 (6)	5.35E-05 (7)	-1.08E-08
2	1.49E-09	3.25E-02	1.49E-09	3.25E-02	0.00E+00
3a	6.33E-08	4.09E-05	6.33E-08	4.09E-05	0.00E+00
3b	1.58E-08	1.02E-04	1.60E-08 (4)	1.03E-04 (5)	1.08E-06
7	7.67E-07	4.90E+00	7.67E-07	4.90E+00	0.00E+00
8	5.97E-08	1.30E+00	5.97E-08	1.30E+00	0.00E+00
Total	**1.74E-06**	**6.24E+00**	**1.74E-06**	**6.24E+00**	**1.06E-06**
Δ Dose	1.10E-04 (17)		1.11E-04 (18)		
CCFP	48.61% (2)		48.62% (12)		
ΔCCFP	0.73% (2)		0.74% (13)		
Class 3b LERF	1.58E-08 (9)		1.60E-08 (10)		**Difference** 1.66E-10 (11)
Delta Class 3b LERF (from 3 in 10 yr base case)	1.27E-08 (14)		1.28E-08 (15)		**Difference** 1.54E-10 (16)
Delta Class 3b LERF (from 1 in 10 yr base case)	5.29E-09 (19)		5.38E-09 (20)		**Difference** 9.52E-11 (21)

Notes:

(1) Values taken from Table 5-4.
(2) Values taken from Table 5-5.
(3) Values for all classes except Class 1 and Class 3b are identical to the “without corrosion” values.
(4) Value calculated by summing “without corrosion” frequency and the ΔF value derived from Eqn. 5-7.
(5) Value obtained by multiplying the Class 3b “with corrosion” frequency calculated in this table and the dose for Class 3b listed in Table 5-3.
(6) Value calculated by using the formula for Class 1 listed in Table 5-3.
(7) Value obtained by multiplying the Class 1 “with corrosion” frequency calculated in this table and the dose for Class 1 listed in Table 5-3.
(8) The values in this column are the difference in dose rate (person-rem/yr) between the “with corrosion” and without corrosion” cases.
(9) Value is identical to the Class 3b “without corrosion” frequency listed above.
(10) Value is identical to the Class 3b “with corrosion” frequency listed above.
(11) Value is the difference in the “with corrosion” and “without corrosion” Class 3b LERF values.
(12) Value calculated based upon Eqn. 5-5.

(13) Value is the difference between the "with corrosion" CCFP in this table minus the "with corrosion" CCFP in Table 5-7.
(14) Value is the difference between the "without corrosion" Class 3b LERF in this table minus the "without corrosion" Class 3b LERF in Table 5-7.
(15) Value is the difference between the "with corrosion" Class 3b LERF in this table minus the "with corrosion" Class 3b LERF in Table 5-7.
(16) Value is the difference in the "with corrosion" and "without corrosion" Delta Class 3b LERF (from 3 in 10 yr case) values.
(17) Value is the difference between the "without corrosion" Dose (person-rem/yr) in this table minus the "without corrosion" Dose in Table 5-7.
(18) Value is the difference between the "with corrosion" Dose (person-rem/yr) in this table minus the "with corrosion" Dose in Table 5-7.
(19) Value is the difference between the "without corrosion" Class 3b LERF in this table minus the "without corrosion" Class 3b LERF in Table 5-8.
(20) Value is the difference between the "with corrosion" Class 3b LERF in this table minus the "with corrosion" Class 3b LERF in Table 5-8.
(21) Value is the difference in the "with corrosion" and "without corrosion" Delta Class 3b LERF (from 1 in 10 yr case) values.

Additional corrosion sensitivity cases were developed to gain an understanding of the sensitivity of the results to the key parameters in the corrosion risk analysis. The time for the flaw likelihood to double was adjusted from every five years to every two and every ten years. The failure probabilities for the cylinder and dome and the basemat were increased and decreased by an order of magnitude. The total detection failure likelihood was adjusted from 10% to 15% and 5%. The results are presented in Table 5-10. In every case, the impact from including the corrosion effects is very minimal. Even the upper bound estimates with very conservative assumptions for all of the key parameters yield increases in LERF due to corrosion of only 4.83E-9/yr. The results indicate that even with very conservative assumptions, the conclusions from the "base case" analysis would not change. The tables in Appendix C show the internals of these calculations which are then summarized in Table 5-10.

Table 5-10 – Summary of Additional Corrosion Sensitivities

Case	Case Description	Age (a)	Containment Breach (b)	Visual Inspection and Non-Visual Flaws (c) (d)	Increase in 3b Frequency (LERF) for ILRT Extension from 3 per 10 years to once per 15 years	
					Total Increase (yr^{-1})	Increase Due to Corrosion (yr^{-1}) (f)
1	Base Case	Doubles every 5 yrs	1% Cylinder 0.1% Basemat	10%	3.08E-10	1.54E-10
2	Age - Upper Bound	Doubles every 2 yrs	Base	Base	4.96E-10	3.42E-10
3	Age - Lower Bound	Doubles every 10 yrs	Base	Base	2.80E-10	1.26E-10
4	Inspection - Upper Bound	Base	Base	15%	3.82E-10	2.28E-10
5	Inspection - Lower Bound	Base	Base	5%	2.47E-10	9.25E-11
6	Breach - Upper Bound	Base	10% Cylinder 1% Basemat	Base	1.70E-09	1.54E-09
7	Breach - Lower Bound	Base	0.1% Cylinder 0.01% Basemat	Base	1.70E-10	1.54E-11
8	Overall Lower Bound (e)	Doubles every 10 yrs	0.1% Cylinder 0.01% Basemat	5%	1.62E-10	7.56E-12
9	Overall Upper Bound	Doubles every 2 yrs	10% Cylinder 1% Basemat	15%	4.94E-09	4.78E-09

Notes
(a) Step 3 in the corrosion analysis
(b) Step 4 in the corrosion analysis
(c) Step 5 in the corrosion analysis
(d) Percentage value is for the "cylinder" portion of the containment (the basemat is assumed to be uninspectable)
(e) Lower Bound case implements a "5% Cylinder 100% Basemat" visual inspection non-accessibility factor; this differs from what is listed in Table 6-1 in Appendix H of Reference 26. This interpretation is consistent with the other sensitivity cases and the fact that the containment basemat is assumed to be uninspectable.
(f) The "Increase Due to Corrosion" refers to the difference between the "once per 15 year" and "3 per 15 year" ILRT intervals; the corrosion parameters associated with each case are identically applied to the two intervals examined.

5.4.2 Potential Impact from External Events

External hazards were evaluated in the Fermi 2 Individual Plant Examination of External Events (IPEEE) submittal (Reference 24) in response to the NRC IPEEE Program (Generic Letter 88-20, Supplement 4 [Reference 19]). The IPEEE program was a one-time review of external hazard risk and was limited in its purpose to the identification of potential plant vulnerabilities and the understanding of associated severe accident risks. The results of the Fermi 2 IPEEE study are documented in the Fermi 2 IPEEE main report. The primary areas of external event evaluation at Fermi 2 were internal fire and seismic.

The internal fire events were addressed by using the Fire-Induced Vulnerability Evaluation (FIVE) methodology (Reference 29). As such, there are no realistic CDF or LERF values available from the IPEEE to support the ILRT extension risk assessment. Currently, a state of the art Fermi 2 Fire PRA model, which will meet appropriate CC-II requirements in the ASME PRA Standard, is being developed. Until the Fire PRA model is complete and peer reviewed, the IPEEE analysis will be used, consistent with NEI 04-10, and the fire risk insights will be complemented by conservative qualitative potential impact of the fire hazard.

In the Fermi 2 IPEEE, the seismic risk evaluation was performed in accordance with EPRI Seismic Margins Analysis (SMA) methodology. Since the SMA approach was used, there are no comprehensive CDF and LERF values available from the seismic analysis in the Fermi 2 IPEEE to support the ILRT extension risk evaluation. A conclusion from the SMA was that Fermi 2 has a high-confidence-low-probability-of-failure (HCLPF) capacity of at least 0.3 peak ground acceleration (PGA). Currently, a state of the art Fermi 2 Seismic PRA model, which will meet appropriate CC-II requirements in the ASME PRA Standard, is being developed. Until the Seismic PRA model is complete and peer reviewed, the IPEEE analysis, in conjunction with other qualitative and quasi-quantitative information, will be used to assess the impact of seismic hazards for risk informed applications.

Fermi 2 performed an evaluation of all external hazards (39 total hazards including internal fire, internal flooding, seismic, high winds, tornados, external floods, transportation accidents, nearby facility accidents, and release of onsite chemicals). The result of this evaluation was that 36 external hazards were screened with respect to CDF and LERF risk. Only internal flooding, internal fire, and seismic activity require detailed PRAs. The risk screening of hazards implies that each hazard has a mean CDF significantly lower than 1E-06/year. This evaluation of all external hazards was reviewed by a peer review team in April 2014 using ASME/ANS RA-Sb-2009 and Regulatory Guide 1.200, Rev 2. All supporting requirements were met to CC-II. Therefore, these screened hazards are not significant dominant contributors to external event risk and since quantitative analysis of these events is not practical, the external event quantitative treatment will be developed based on seismic and internal fire risk. It should be noted that the internal flooding hazard is included in the quantitative solution to the Fermi 2 internal events model.

Since the seismic and internal fire PRA models at Fermi 2 are in development (and, as such, cannot be used to support risk-informed applications), it is necessary to develop a

multiplier that can be applied to the internal events PRA results to account for the risk contribution from non-screened external events in the ILRT evaluation.

Fermi 2 used a SMA method in the IPEEE to address seismic risk and thus no seismic core damage estimate was developed. However, there is a relatively current estimate for the seismic risk for Fermi 2 which was developed by the NRC as part of its work to address Generic Issue 199 (GI-199, Reference 30), *Implications of Updated Probabilistic Seismic Hazard Estimates in Central and Eastern U.S. for Existing Plants*. These results are provided in a safety/risk assessment that the NRC performed for addressing Generic Issue 199. This assessment determined that the weakest link model seismic risk for Fermi 2 is 4.2E-06 per year. It should be noted that the RAI responses for the SAMA application used a frequency of 2.26E-06/yr based upon updated hazard curves; however, the 4.2E-06/yr will be used in the ILRT application to ensure a conservative bias to the calculations is applied.

The conclusion of the Fermi 2 FIVE analysis in the IPEEE was that all rooms were screened from further consideration and there are no fire-induced vulnerabilities associated with the continued operation of the Fermi 2. However, the core damage estimates for the areas in the final phase of screening are typically used to represent the fire risk. Table 4-1 provides a listing of those areas and their associated CDF which totals to 2.15E-05/yr. The EPRI FIVE methodology results are conservative and are not comparable to internal events core damage frequencies. This is especially true when considering that the original FIVE analysis used the Fermi 2 IPE as the basis for the core damage assessments used in FIVE. The Fermi 2 PRA model has been updated many times since the IPE model and the current internal events CDF is 1.7E-06/Rx-yr compared to the IPE CDF of 5.7E-06/Rx-yr. This is approximately a factor of four less than original IPE CDF and it could be reasonably assumed that an update of the FIVE analysis with this model would result in a fire CDF of 5.4E-06/Rx-yr, one-fourth of the original fire CDF. This would account for updated modeling of the internal events portion of the model that was used in the FIVE analysis, but not necessarily address all of the conservatisms inherently to the FIVE methodology. Even though a larger reduction in the Fire CDF may be justifiable, the Fermi 2 fire CDF has been conservatively reduced by a factor of two to 1.08E-05/Rx-yr for the SAMA analysis. This is well within the range suggested in the NEI 05-01.

Therefore, the external event multiplier (EEM) for Fermi 2 is determined as follows:

EEM = (Internal Event CDF + Seismic CDF + Fire CDF) / Internal Event CDF (Eqn. 5-8)

= (1.7E-06 + 4.2E-06 + 1.08E-05)/1.7E-06 = 9.8

This result is approximately equal to a value of 10. Therefore, an external event multiplier of 10 is used to determine external event impact for the ILRT analysis. It should be noted that based upon the formulation for the EEM used in Eqn. 5-8, the EEM is applied to the internal events LERF results in Table 5-11, Table 5-12, and Table 5-13 to derive the "combined" impact (the external events contribution is derived by subtracting the "Combined" result from the "Internal Events" result).

Table 5-11 provides a summary of the combined impact of internal and external events on the increase in LERF for various ILRT intervals.

Table 5-11 – Impact of External Events on ILRT LERF (EEM = 10)

Hazard	3 per 10 years	1 per 10 years	1 per 15 years	LERF Increase (vs 3 in 10 years)	LERF Increase (vs 1 in 10 years)
Internal Events	3.16E-09	1.05E-08	1.58E-08	1.27E-08	5.29E-09
External Events	2.85E-08	9.48E-08	1.42E-07	1.14E-07	4.76E-08
Combined	3.16E-08	1.05E-07	1.58E-07	1.27E-07	5.29E-08

Since there is a significant amount of uncertainty associated with the determination of the external event multiplier, it was deemed appropriate to evaluate the impact of the EEM were increased by 50% (to a numerical value of 15); Table 5-12 summarizes this evaluation.

Table 5-12 – Impact of External Events on ILRT LERF (EEM = 15)

Hazard	3 per 10 years	1 per 10 years	1 per 15 years	LERF Increase (vs 3 in 10 years)	LERF Increase (vs 1 in 10 years)
Internal Events	3.16E-09	1.05E-08	1.58E-08	1.27E-08	5.29E-09
External Events	4.43E-08	1.48E-07	2.22E-07	1.77E-07	7.40E-08
Combined	4.75E-08	1.58E-07	2.37E-07	1.90E-07	7.93E-08

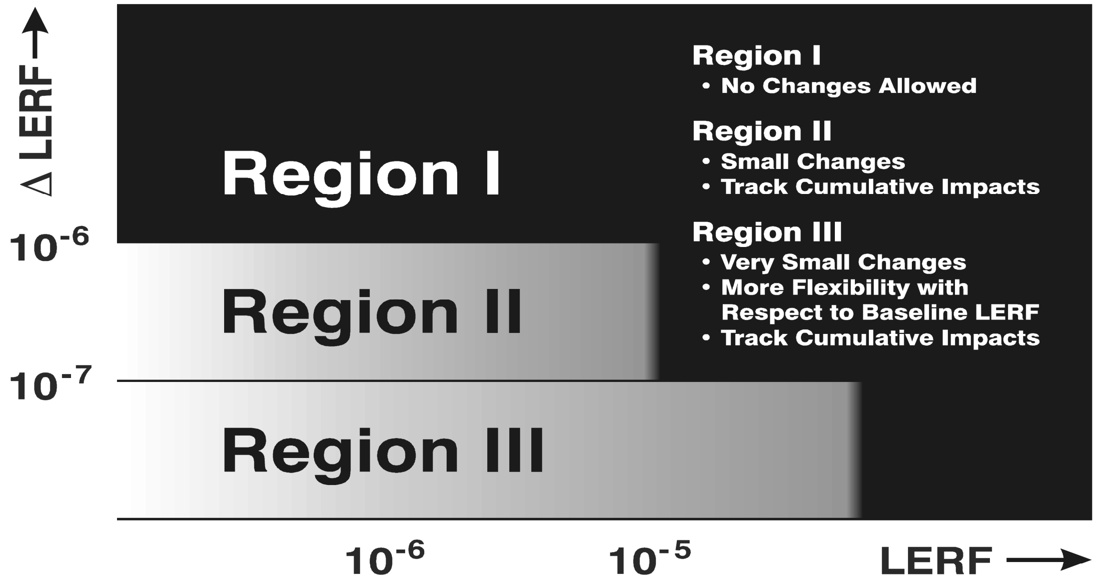
Since the LERF increase for both an EEM of 15 and an EEM of 10 is greater than 1.0E-7 for the 15 year extension increase (as compared to the base case), it is necessary to tabulate the baseline LERF values for the purpose of comparisons with RG 1.174 thresholds. Table 5-13 presents those values.

Table 5-13 – Baseline LERF Values

Parameter	Frequency (yr^{-1})
LERF-Internal	5.25E-07
LERF-External-10X	4.73E-06
LERF-Combined-10X	5.25E-06
LERF-External-15X	7.35E-06
LERF-Combined-15X	7.88E-06

RG 1.174 provides NRC recommendations for using risk information in support of applications requesting changes to the license basis of the plant. The Regulatory Guide 1.174 acceptance guidelines are used here to assess the ILRT interval extension. Those acceptance guidelines are presented in Figure 5-1 (which is excerpted from RG 1.174).

Figure 5-1 – RG 1.174 Acceptance Guidelines for LERF



Using the bounding case (EEM=15), the calculated increase in LERF due to the combined internal and external events from extending the ILRT testing frequency from three per 10 years to one per 15 years is 1.90E-7/yr. Per Regulatory Guide 1.174, this is a "small change" (increase) in LERF given that the baseline value of 7.88E-6/yr for this bounding case is between 1E-6/yr and 1E-5/yr.

5.4.3 Impact of Credit for Containment Overpressure

Section 1.2 of Reference 26 states the following regarding containment overpressure:

> *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.*

Since containment overpressure is not required in the Fermi 2 design basis to support ECCS performance to mitigate accidents at Fermi 2, the effect of the ILRT extension on containment overpressure need not be addressed. However, since the Fermi 2 PRA model credits overpressure in long-term loss of containment heat removal scenarios, a sensitivity study has been performed which addresses this topic. It should be noted that the Fermi 2 PRA model does not credit overpressure in LOCA scenarios where the initial containment pressurization helps to satisfy the NPSH requirements for early injection.

The following analytical assumptions ensure that this sensitivity study contains a strong conservative bias.

1. It is assumed that only the EPRI Class 3b containment leakage terms (see Table 5-1) contribute to a loss of overpressure and, as a result, the loss of NPSH (this is consistent with the guidance in Reference 26).
2. No credit is taken for the beneficial effects of the reduction in containment pressure due to the postulated leakage. Crediting this leakage in scenarios where loss of NPSH is postulated would model the absence of SRV reclosure at high containment pressures; crediting this leakage would also allow credit for low pressure injection from external sources (such as RHRSW) even if containment venting is not performed.
3. It is assumed that the containment leakage will result in ECCS equipment failure probabilities identical to those of when containment venting occurs to the third floor of the reactor building (as opposed to the venting stack or the fifth floor of the reactor building). This is conservative, since leakage high in the drywell (e.g. at the drywell head seal) would not cause this effect.
4. No modification was made to the human error probability for establishing long-term torus cooling; this is an easily performed, EOP directed action. The HEP is insensitive to variations in available time which are realistic for the scenario in question.

To perform this sensitivity study, it was necessary to modify the baseline Fermi 2 PRA model to incorporate model logic to initiate the failure mode. Appendix B discusses this modification in detail.

The purpose of the sensitivity is to examine the increase in CDF (delta-CDF) and the increase in LERF (delta-LERF) that occur when the ILRT frequency is increased as determined by the equations below:

$$\text{delta-CDF} = CDF_{OP\text{-}1in15} - CDF_{OP\text{-}3in10} \qquad \text{(Eqn. 5-9)}$$

$$\text{delta-LERF} = LERF_{OP\text{-}1in15} - LERF_{OP\text{-}3in10} \qquad \text{(Eqn. 5-10)}$$

Where,

$CDF_{OP\text{-}1in15}$ is the Core Damage Frequency associated with performing the ILRT once per 15 years,
$CDF_{OP\text{-}3in10}$ is the Core Damage Frequency associated with performing the ILRT three times per 10 years,
$LERF_{OP\text{-}1in15}$ is the Large Early Release Frequency associated with performing the ILRT once per 15 years, and
$LERF_{OP\text{-}3in10}$ is the Large Early Release Frequency associated with performing the ILRT three times per 10 years,

The results of the sensitivity study are presented below in Table 5-14.

Table 5-14 – Overpressure Sensitivity Results

Parameter	Frequency(yr^{-1})
$CDF_{OP\text{-}3in10}$	1.76E-06
$CDF_{OP\text{-}1in15}$	1.81E-06
delta-CDF	5.25E-08
$LERF_{OP\text{-}3in10}$	5.16E-07
$LERF_{OP\text{-}1in15}$	5.51E-07
delta-LERF	3.54E-08

The above results show that, since the delta-CDF value is substantially below 1.0E-6/yr, the increase in this metric may be classified as being "very small" by the criteria outlined in RG 1.174. The above results also show that, since the delta-LERF value is substantially below 1.0E-7/yr, the increase in this metric may be classified as being "very small" by the criteria outlined in RG 1.174. These results in the context of the substantial analytical conservatisms in the calculations, demonstrate that containment overpressure has a minimal effect on the risk associated with permanently extending the ILRT interval to one in 15 years.

The above sensitivity results (and the substantial conservatism in the calculations) justify the screening of the CDF impact from further consideration.

6. Results

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

Table 6-1 – ILRT Extension Summary

Class	Dose (person-rem)	Base Case 3 in 10 years		Base Case 1 in 10 years		Base Case 1 in 15 years	
		Freq [yr^{-1}]	Person-rem/yr	Freq [yr^{-1}]	Person-rem/yr	Freq [yr^{-1}]	Person-rem/yr
1	6.46E+01	8.92E-07	5.76E-05	8.55E-07	5.53E-05	8.29E-07	5.35E-05
2	2.18E+07	1.49E-09	3.25E-02	1.49E-09	3.25E-02	1.49E-09	3.25E-02
3a	6.46E+02	1.27E-08	8.18E-06	4.22E-08	2.73E-05	6.33E-08	4.09E-05
3b	6.46E+03	3.16E-09	2.04E-05	1.05E-08	6.81E-05	1.58E-08	1.02E-04
7	6.39E+06	7.67E-07	4.90E+00	7.67E-07	4.90E+00	7.67E-07	4.90E+00
8	2.18E+07	5.97E-08	1.30E+00	5.97E-08	1.30E+00	5.97E-08	1.30E+00
Total	N/A	1.74E-06	6.24E+00	1.74E-06	6.24E+00	1.74E-06	6.24E+00
ILRT Dose Rate from 3a and 3b							
ΔTotal Dose Rate	From 3 Years	N/A		6.68E-05 (1)		1.14E-04 (2)	
	From 10 Years	N/A		N/A		4.77E-05 (3)	
Δ % Dose Rate	From 3 Years	N/A		0.00107% (4)		0.00184% (5)	
	From 10 Years	N/A		N/A		0.00076% (6)	
3B Frequency LERF							
ΔLERF	From 3 Years	N/A		7.37E-09 (7)		1.27E-08 (7)	
	From 10 Years	N/A		N/A		5.29E-09 (8)	
CCFP%							
ΔCCFP (%)	From 3 Years	N/A		0.42% (7)		0.73% (7)	
	From 10 Years	N/A		N/A		0.30% (9)	

Notes:

(1) Values based upon the difference between the sum of the dose rates (in person-rem/yr) for Class 3a and Class 3b for the "1 in 10 year" case minus the sum of the dose rates for Class 3a and Class 3b for the "3 in 10 year" case.

(2) Values based upon the difference between the sum of the dose rates (in person-rem/yr) for Class 3a and Class 3b for the "1 in 15 year" case minus the sum of the dose rates for Class 3a and Class 3b for the "3 in 10 year" case.

(3) Values based upon the difference between the sum of the dose rates (in person-rem/yr) for Class 3a and Class 3b for the "1 in 15 year" case minus the sum of the dose rates for Class 3a and Class 3b for the "1 in 10 year" case.
(4) Values based upon the formula [ΔTotal Dose Rate$_{1in10yr}$ / Dose Rate$_{3in10yr}$].
(5) Values based upon the formula [ΔTotal Dose Rate$_{1in15yr}$ / Dose Rate$_{3in10yr}$].
(6) Values based upon the formula [ΔTotal Dose Rate$_{1in15yr}$ / Dose Rate$_{1in10yr}$].
(7) Values taken from Table 5-5.
(8) Value calculated from inputs in this table based upon the formula [Freq$_{3B-1in15}$ - Freq$_{3B-1in10}$].
(9) Value calculated from information in Table 5-5 based upon the formula [CCFP$_{1in15}$ – CCFP$_{1in10}$].

7. <u>Conclusions</u>

Based on the results from Section 5 and the sensitivity calculations presented in Section 6, the following conclusions regarding the assessment of the plant risk are associated with extending the Type A ILRT test frequency to fifteen years:

1. Reg. Guide 1.174 (Reference 4) provides guidance for determining the risk impact of plant-specific changes to the licensing basis. Reg. Guide 1.174 defines very small changes in risk as resulting in increases of CDF below 10^{-6}/yr and increases in LERF below 10^{-7}/yr. Since the ILRT does not impact CDF, the relevant criterion is LERF. The increase in LERF resulting from a change in the Type A ILRT test interval from three in ten years to one in fifteen years is very conservatively estimated as 1.27E-08/yr using the EPRI guidance as written,. As such, the estimated change in LERF is determined to be "very small" using the acceptance guidelines of Reg. Guide 1.174.

2. Regulatory Guide 1.174 (Reference 4) also states that when the calculated increase in LERF is in the range of 1.0E-06 per reactor year to 1.0E-07 per reactor year, applications will be considered only if it can be reasonably shown that the total LERF is less than 1.0E-05 per reactor year. An additional assessment of the impact from external events was also made. In this case, the total LERF increase was conservatively estimated (with an external event multiplier of 15) as 1.90E-07 for Fermi 2 (the baseline total LERF for this case is 7.88E-06/yr). This is well below the RG 1.174 acceptance criteria for total LERF of 1.0E-05.

3. The change in Type A test frequency to once-per-fifteen-years, measured as an increase to the total integrated plant risk for those accident sequences influenced by Type A testing, is 1.14E-4 person-rem/yr (a 0.00184% increase). EPRI Report No. 1009325, Revision 2-A 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. Moreover, the risk impact when compared to other severe accident risks is negligible.

4. The increase in the conditional containment failure frequency from the three in ten year interval to one in fifteen year interval is 0.73%. EPRI Report No. 1009325, Revision 2-A states that increases in CCFP of less than or equal to 1.5 percentage points are very small. Therefore this increase judged to be very small.

5. A sensitivity case was performed which examined the effect of containment overpressure on CDF and LERF; it should be noted that containment overpressure is not credited in the Fermi design basis. Substantial conservatism was included in the sensitivity to ensure that it was bounding. The increase in CDF for this sensitivity was 5.25E-08/yr and the increase in LERF was 3.54E-08/yr. The estimated change in both CDF and LERF was determined to be "very small" using the acceptance guidelines of Reg. Guide 1.174. This sensitivity was utilized to justify the screening of the CDF impact from further consideration.

Therefore, increasing the ILRT interval to once per 15 years is considered to be insignificant, since it represents a very small change to the Fermi 2 risk profile.

8. References

The following is a list of references that were used in this guidance document. It should be noted that several of the references are not "called out" in the text of this evaluation; however, to maintain consistency with the reference numbers in the template provided in Reference 26, these references were retained in the document and can be viewed as being bibliographic in nature.

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3. *Interim Guidance for Performing Risk Impact Assessments In Support of One-Time Extensions for Containment Integrated Leakage Rate Test Surveillance Intervals*, Rev. 4, Developed for NEI by EPRI and Data Systems and Solutions, November 2001.
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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.
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8. Letter from R. J. Barrett (Entergy) to U.S. Nuclear Regulatory Commission, IPN-01-007, January 18, 2001.
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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.

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16. United States Nuclear Regulatory Commission, Reactor Safety Study, WASH-1400, October 1975.
17. *Release of the FermiV10 PRA Model*, TMSA-15-0040, November 23, 2015.
18. *Fermi 2 Component Data Notebook, Vol. 1*, EF2-PRA-010 Rev 1, March 2013.
19. *Individual Plant Examination of External Events (IPEEE) for Severe Accident Vulnerabilities - 10 CFR 50.54(f), Supplement 4*, NRC Generic Letter 88-20, June 1991.
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.
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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. *Risk Impact Assessment of Extended Integrated Leak Rate Testing Intervals,* EPRI, Palo Alto, CA: 2007. 1009325 Rev 2-A.
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28. *Fermi 2 License Renewal Application*, DTE Energy Company, April 2014.
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32. *Standard for Probabilistic Risk Assessment for Nuclear Power Plant Applications*, ASME/ANS RA-Sa-2009 Addendum A of ASME RA-S-2008, February 2009.
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34. *Enrico Fermi 2 Nuclear Power Plant Focused Scope PRA Peer Review Report Using ASME/ANS PRA Standard Requirements - Final Report*, Boiling Water Reactor Owner Group (BWROG), April 2014.
35. *Fermi 2 Nuclear Power Station Other External Hazards PRA Peer Review Report Using ASME/ANS PRA Standard Requirements - Final Report*, Boiling Water Reactor Owner Group (BWROG), June 2014.
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37. *Process for Performing PRA Peer Reviews Using the ASME PRA Standard (Internal Events)*, NEI 05-04 Revision 2, November 2008.
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39. Fermi 2 Surveillance Performance Forms 43.401.100.071110 and 43.401.100.921101.

A. <u>Appendix A – PRA Model Technical Adequacy</u>

The FermiV10 model update is the most recent evaluation of the risk profile at Fermi 2 for internal event (including internal flooding) challenges. The Fermi 2 PRA modeling is highly detailed, including a wide variety of initiating events, modeled systems, operator actions, and common cause events. The PRA model quantification process used for the Fermi 2 PRA is based on the event tree/fault tree methodology.

Fermi 2 employs a multi-faceted approach for establishing and maintaining the technical adequacy and plant fidelity of the PRA model. This approach includes both a proceduralized PRA maintenance and update process and the use of independent peer reviews. The following information describes this approach as it applies to the Fermi 2 PRA.

A.1 PRA Maintenance and Update

The Fermi 2 risk management process ensures that the PRA model remains an accurate reflection of the as-built and as-operated plant. This process is defined in the Fermi 2 PRA model maintenance and configuration control program in accordance with the governing procedure. The procedure delineates the responsibilities and guidelines for updating the full power internal events PRA model. It also defines the process for implementing regularly scheduled and interim PRA model updates, for tracking issues identified as potentially affecting the PRA models (e.g., due to changes in the plant, errors or limitations identified in the model, industry operating experience), and for controlling the model and associated computer files. To ensure the current PRA model remains an accurate reflection of the as-built, as-operated plants, the following activities are routinely performed:

- Design changes and procedure changes are reviewed for their impact on the PRA model.
- Maintenance unavailabilities are captured, and their impact on Core Damage Frequency (CDF) is trended.
- Plant specific initiating event frequencies, failure rates, and maintenance unavailabilities are updated approximately every three years. Longer intervals may be justified if it can be shown that the PRA continues to adequately represent the as-built, as-operated plant. Section A.3 shows the brief history of the major Fermi 2 PRA model updates.

In addition to these activities, Fermi 2 risk management procedures provide the guidance for particular risk management, PRA quality, and maintenance activities. This guidance includes:

- Documentation of the PRA model, PRA products, and bases documents.
- The approach for controlling electronic storage of risk management products including PRA update information, PRA models, and PRA applications.
- Guidelines for updating the full power, internal events PRA models for Fermi 2.
- Guidance for use of quantitative and qualitative risk models in support of the On-Line Work Control Process Program for risk evaluations for maintenance tasks (corrective

maintenance, preventive maintenance, minor maintenance, surveillance tests and modifications) on systems, structures, and components (SSCs) within the scope of the Maintenance Rule (10CFR50.65(a)(4)).

A.2 Plant Changes not yet incorporated into the PRA Model

A review of plant modifications and procedure changes was performed following the plant refueling outage which concluded in November 2015. This review concluded that there were several modifications and procedures with PRA impact (associated with implementation of the Fermi 2 FLEX strategy), that were not incorporated into the PRA model. However, since these changes added additional mitigating capabilities for long term loss offsite power scenarios, the current PRA model is conservative with respect to the current plant configuration and also with respect to the ILRT extension request calculations.

A.3 Fermi 2 PRA Model Update History

Table A-1 shows a brief history of the Fermi 2 PRA model.

Table A-1 – Fermi 2 PRA Model History

Model (Year Issued)	Description	CDF (Per Yr)	LERF (Per Yr)	Includes Internal Flooding? (2)
PLG-0676 (1989)	Original Fermi 2 PRA.	2.2E-05	NA	No
IPE (1992)	Model developed in response to NRC Generic Letter 88-20.	5.7E-06	8.0E-07	No
PSA97C (1997)	RISKMAN model which was reviewed using the NEI Peer Review process.	7.1E-06	1.2E-06	No
FermiV2 (2002)	CDF Model converted from RISKMAN to CAFTA.	5.0E-06	N/A	No
FermiV3 (2002)	Normal PRA Maintenance and CAFTA Level 2 Model developed.	3.3E-06	2.5E-07	No
FermiV4 (2003)	Model updated as part of normal PRA Maintenance.	5.8E-06	9.3E-07	Yes(3)
FermiV5 (2004)	Model updated was part of the Extended Power Uprate (EPU) evaluation (Model was not issued)(1).	N/A	N/A	Yes(3)
FermiV6 (2004)	Model updated as part of normal PRA Maintenance.	6.1E-06	4.8E-07	Yes(3)

Model (Year Issued)	Description	CDF (Per Yr)	LERF (Per Yr)	Includes Internal Flooding? (2)
FermiV7 (2006)	Model updated to close all A and B NEI Peer Review Findings & Observations which may have impact Mitigating Systems Performance Index (MSPI) results.	1.4E-05	5.5E-07	Yes(3)
FermiV8 (2010)	Periodic update to incorporate accident sequence changes to improve MSPI margin and to address the backlog of identified issues in the modeling database.	2.3E-06	3.1E-07	Yes(3)
FermiV9 (2013)	Complete model upgrade including Initiating Events, Success criteria, Data, System Notebooks, HRA, Internal Flood, MAAP 4.0.7 Analyses, and Level 2/LERF.	1.5E-06	3.7E-07	Yes
FermiV10 (2015)	Update the HRA Dependency Analysis to a different methodology.	1.7E-06	3.6E-07	Yes

Notes:

(1) EPU was not implemented at Fermi 2.
(2) Includes Internal Flood Initiating Events.
(3) Limited scope internal flooding model based on PLG analysis.

In addition to the model history provided in that table, a brief narrative is presented to clarify the recent model history in relation to the peer reviews to the ASME/ANS PRA Standard (Reference 32) which were performed in recent years.

1. February 2010: PRA model version FermiV8 was released in February 2010. This model version had no peer review performed against the ASME/ANS standard.

2. February 2010 – August 2012: A project was started to upgrade the full power internal events model to meet the Regulatory Guide 1.200, Revision 1, requirements. This project was a PRA upgrade as defined in Section 1-5.4 of the PRA standard for all model aspects including initiating events, success criteria, component data, system modeling, HRA, internal flooding, and LERF. The resulting model was identified as the draft FermiV9 model.

3. August 2012: A peer review was performed in August 2012 on the draft FermiV9 model. This peer review was a complete review of all supporting requirements of the ASME/ANS standard to review all upgrades. All F&Os from this peer review of the draft FermiV9 model that resulted in SRs not meeting Capability Category II requirements were provided in Table 2 of Reference 33.

4. April 2013: PRA model version FermiV9 was released in April 2013. This release provided resolution to the F&Os identified by the peer review team in August 2012 on the draft FermiV9 model. No PRA upgrades were instituted into the FermiV9 model following the August 2012 peer review. All changes made to the model to resolve the F&Os identified in Table 2 of Reference 33 were defined as PRA maintenance because there were no new methodologies or significant changes in scope or capability that affected the significant accident sequences. Therefore, no further peer reviews were required on the FermiV9 model.

5. February 2014: Following release of FermiV9 in April 2013, the HRA dependency analysis was re-performed to support modeling of the Fermi 2 fire PRA and seismic PRA and incorporated into a draft FermiV10 model. This HRA dependency analysis was considered an upgrade per the PRA standard because a new methodology was instituted. In February 2014, a focused scope peer review of the draft FermiV10 model HRA dependency analysis was performed. This peer review concluded that all reviewed supporting requirements met Capability Category II.

6. November 2015: The FermiV10 model was released as the model of record. This is essentially the same model with respect to the CDF and LERF end states which was peer reviewed in the February 2014 focused scope peer review. The one finding that emanated from the February 2014 focused scope peer review on the HRA dependency analysis was addressed in this model release.

Based on the above timeline, the following observations are made:

1. There were many changes made between FermiV8 and the draft FermiV9 model. These changes included new methodologies, and significant changes in scope and capability that impacted the significant accident sequences. Changes included re-evaluation of initiating events, success criteria, component data, system modeling, HRA, internal flooding, and LERF to current industry standards.
2. DTE considered all changes incorporated into the draft FermiV9 model as PRA upgrades as defined in Section 1-5.4 of the ASME/ANS PRA standard as there were significant changes that impacted the significant accident sequences and accident progression sequences.
3. A peer review was performed in August 2012 on the draft FermiV9. This peer review was a complete review of all supporting requirements of the ASME/ANS standard to review all upgrades. All F&Os that did not meet Capability Category II from this peer review of the draft FermiV9 model were provided in Table 2 of Reference 33.
4. All findings from the August 2012 peer review of the draft FermiV9 model were addressed in the FermiV9 model released in April 2013. All changes made to the draft FermiV9 model were defined as PRA maintenance, because there were no new methodologies or significant changes in scope or capability that affected the significant accident sequences.

5. The focused scope peer review in February 2014 (Reference 35) superseded the August 2012 peer review for the supporting requirements (SRs) pertaining to the HRA dependency analysis. Therefore, the findings related to HRA dependency from the 2012 Peer Review need not be tracked for the FermiV10 model of record (nonetheless, the resolution to these items is discussed in Section A.4).

A.4 Applicability of Peer Review Findings and Observations

Several assessments of technical capability have been made, and continue to be planned for the Fermi 2 PRA model. For the current model of record the following assessments were performed and are discussed in the paragraphs below:

1. In August 2012, a peer review was held at the Fermi 2 site under the auspices of the Boiling Water Reactor Owners Group (BWROG), using the NEI 05-04 PRA Peer Review process (Reference 37), the ASME PRA Standard ASME/ANS RA-Sa-2009 (Reference 32) and Regulatory Guide 1.200, Rev 2 (Reference 36). The 2012 Fermi 2 PRA Peer Review was a full-scope review of all the Technical Elements of the internal events, at-power PRA. The BWROG peer review final report was issued in November 2012 (Reference 33). All open and closed gaps to meet capability category II of the ASME/ANS PRA Standard are identified in Table A-2. A PRA model update was started following issuance of the peer review final report and was completed in April 2013. This update (FermiV9) addressed the gaps described in the peer review. Table A-2 contains the actions taken to resolve all gaps.
2. The Human Reliability Analysis (HRA) dependency analysis was updated in 2013. Because a different methodology was used to perform the HRA dependency analysis, a focused scope peer review of the HRA dependency analysis was performed using ASME/ANS RA-Sb-2009 and Regulatory Guide 1.200, Rev 2 in February 2014. The peer review team found that the dependency analysis met Capability Category II (CC-II) for all evaluated supporting requirements (Reference 34). Three findings listed in Table A-2 from the August 2012 Peer Review were superseded by the re-review in the focused scope peer review for the HRA dependency analysis of the technical elements associated with those findings.
3. In April 2014, a focused scope peer review of Other External Hazards (not including internal fire and seismic) was performed using ASME/ANS RA-Sb-2009 and Regulatory Guide 1.200, Rev 2. The peer review team found that the external hazards analysis met CC-II for all evaluated supporting requirements (issued report is listed as Reference 35).

Table A-2 – Resolution of Fermi 2 Internal Events Peer Review F&Os not Meeting Category II

F&O / Status / SR Capability Category	Finding	Resolution
1-22 / Closed SR QU-C2 Not Met	The current approach of using a single such event that is applied to most post-initiator and recovery HFEs, using a single joint probability, while probably generally conservative, may introduce non-conservatisms in specific cases. The probabilities of the events included in the HE1D-D-OPERATOR event range from the E-2 range down to the E-4 range. A cutset containing only two E-2 HFEs would be underestimated by the 1E-6 global value. Additional sensitivity studies should be performed. The Uncertainty notebook includes a sensitivity study with all HEPs increased to the 95 percentile, which results in a factor of 3 increase in CDF. However, the assigned error factor for HE1D-D-OPERATOR is set to 10, which may understate its uncertainty.	"Not Met" classification for SR superseded by 2014 focused scope peer review on HRA dependency; the SR was found to be "Met" to CC II in the focused scope Peer Review.
1-26 / Addressed SR IEFV-A7 Not Met	As noted in section 2.2.9.1 of EF2-PRA-012, maintenance-induced floods were not included on the basis of the fact that only a few minor floods occurred over the past few years. However, past history has shown that significant floods can occur due to maintenance errors, especially on large volume systems such as circulating water, fire protection, condensate, etc. Historical data (as tabulated in Appendix H) confirms the existence of such events, although none have recently occurred. The Fermi 2 internal flooding PRA should consider maintenance-induced flood events on at least the large water volume systems.	Supporting requirement IFEV-A7 is considered to be met because generic data was considered in the evaluation (as revised by the information presented in the resolution to this finding). The evaluation is complete and has been added to the internal flood analysis under the discussion of maintenance induced floods. The large Circ. Water failure flooding the Turbine Building is increased by 1E-3/Rx Yr to reflect this calculated maintenance induced failure frequency. As maintenance induced floods have been evaluated and the frequencies updated in the model, this SR meets Capability Category II.

F&O / Status / SR Capability Category	Finding	Resolution
2-16 / Closed SR HR-G7 Not Met	It was noted that an HEP dependency analysis was performed. However, in Section 5.3.2.2 of Fermi 2 HRA Notebook (EF2-PRA-004), it is stated that the chronological sequencing of HEPs is not used as a criterion in the dependency quantification. SR HR-G7 indicates however that the dependency analysis must account for the influence of success or failure in preceding human actions and system performance on the human event under consideration. Therefore the order in which the operator is presented with opportunities in an accident sequence is important and must be considered. Although it was stated that the chronology of the events is not known with precision when modeling groups of events, which is the approach taken by the Fermi 2 HRA analysis, the order in which the HFEs occur in any one cut set or scenario should be apparent. Any alternative approaches used that vary from industry standards must be documented and studies performed to demonstrate the appropriateness of the approach used.	"Not Met" classification for SR superseded by 2014 focused scope peer review on HRA dependency; the SR was found to be "Met" to CC II in the focused scope Peer Review.
3-15 / Addressed SR DA-D4 Capability Category I	There was no formal examination of the Bayesian posterior values for reasonableness. Therefore, this SR is not met. (DA-D4)	The discussion is inserted into the Data Notebook describing the review of the data posteriors after the Bayesian update. It should be noted that a reasonability check was performed as part of the review of the Component Data analysis prior to the Peer Review; however, there was nothing in the documentation stating that this reasonability check had been performed. As the review of data posteriors is now included in the Data Notebook, this SR meets Capability Category II.
3-28 / Closed SR HR-G7 Not Met	It was noted that for the evaluation of the group of HEPs in Section D.3.2.3 of the Fermi 2 HRA Notebook (EF2-PRA-004), the use of the event HEIFRXPCHSML (a steam LOCA) to represent cutsets in which HEIFRXPCHWML (a water LOCA) was non-conservative. The steam LOCA HEP is 1.0E-3, compared to 4.6E-2 for the water LOCA.	"Not Met" classification for SR superseded by 2014 focused scope peer review on HRA dependency; the SR was found to be "Met" to CC II in the focused scope Peer Review.

F&O / Status / SR Capability Category	Finding	Resolution
4-9 / Addressed SR IE-A7 Not Met	To address item SR IE-A7 (a), section 2.4 and Appendix I include industry Operating Experience summaries that include some low power and shutdown events, but may not include all events that occurred at Fermi 2 (e.g., if they were not significant enough to warrant inclusion in industry databases). Appendix J also considers some shutdown events for at-power applicability, but this appendix is based on generic industry initiator lists and does not consider Fermi 2-specific experience. While operations and system engineer interviews that were performed for the SY and HR notebook development tasks asked about potential initiators, the responses are probably limited to only those systems modeled in the PRA and those initiators that may be related to the specific operator actions being investigated. While the above items provide some review of the items required by this SR, these interviews/reviews were conducted for different purposes than this SR addresses. Hence, the requirements for this SR are only partially met.	To address the question with this finding regarding plant-specific initiators in shutdown (or during low power operation) that are applicable to power operation, a search was performed for such events at Fermi 2. Based upon this search, an event during low power operation (see LER 2007-002) was analyzed and conservatively included as a plant-specific turbine trip event in the Bayesian update process for the Turbine Trip (%TX) initiator. The IE frequency for this initiator was adjusted based upon this information (a very minor change in the mean value occurred). Documentation changes were incorporated into the IE Notebook. As an evaluation of shutdown and lower power events has been completed and the results incorporated into the model, this SR meets Capability Category II.
4-16 / Addressed SR QU-D4 Capability Category I	Section 4.6 of the Quantification Notebook, EF2-PRA-013, provides a comparison of CDF and accident class to other BWR plants. However, this comparison fails to explain why the CDF at Fermi 2 is less than or equal to half the CDF of all of the other plants. In addition, there is no breakdown of how the various initiators compare to the other plants such as turbine trip, loss of condenser, etc. that could be used to explain where the major reductions in CDF at Fermi 2 come from and why they are appropriate.	The Quantification Notebook was revised to reference the comparison of the results from a similar plant included in the Uncertainty Analysis Notebook and to explicitly discuss the significant differences. As this comparison is documented in the Quantification Notebook and the differences are explained, this SR meets Capability Category II.

F&O / Status / SR Capability Category	Finding	Resolution
4-21 / Addressed SR IFSN-A6 Not Met	Section 2.2.5 of the Internal Flood Analysis Notebook, EF2-PRA-012, credits the analysis done in the UFSAR to justify not assessing component damage from missiles, pipe whip, and the jet force of fluid discharge for safety-related systems, but does not address the effect of those events on non-safety systems. Section B.2.1 states that the effects of humidity, condensation, temperature, pipe whip, and jet impingement on equipment operability are assessed to be non-significant impacts based on section 2.2. Since the quoted section of the UFSAR did not address humidity, condensation, or temperature and did not consider jet impingement or pipe whip for non-safety systems, the basis for neglecting the effects does not appear to be valid.	The listed mechanisms were assessed qualitatively and were found not to contribute to the Reactor Building flooding events because of the equipment qualification program at Fermi 2. The Auxiliary Building internal flooding scenarios (with the exception of those emanating from the RBCCW Room, which is considered to be part of the Turbine Building for the purpose of this discussion and these emanating from the HPCI/CRD Pump Room which are considered to be part of the Reactor Building for the purpose of this discussion) involve low pressure, low temperature systems that do not pose challenges to other systems due to pipe whip, jet impingement, or high humidity. Therefore, these considerations are not relevant for scenarios in that building. For equipment in the Turbine Building, the conservative assessment is included to assume failure of all equipment in the building given a failure associated with the specified mechanisms. As the flooding effects on non-safety related equipment have been evaluated and the results updated in the model, this SR meets Capability Category II.
4-22 / Addressed SR IFSN-A7 Not Met	In Section 2.2.7 of the Internal Flood Analysis Notebook, EF2-PRA-012, and Section 7.3.2.2.9 of the UFSAR is referenced to state that MOVs outside the containment have weatherproof type enclosures. Section 7.3.2 of the UFSAR covers the Containment and Reactor Vessel Isolation Control System (CRVICS), and sub-section 7.3.2.2.9 discusses CRVICS valves. The statement in the UFSAR is not a global statement about all MOVs in the plant. In the Internal Flood Walkdown Summary Notebook, EF2-PRA-011, picture 251 shows an MOV in the plant that does not appear to be inside a weatherproof enclosure. The rationale for screening MOVs from spray effects does not appear to be valid. Picture 248 in the IF Walkdown Notebook shows two AOV and SOVs which also do not appear to be protected from spray.	The treatment of MOVs and other components with respect to spray in the internal flood model includes several layers of investigation: • Walkdown evaluation • Use of design and deterministic criteria for Reactor Building Equipment • Comparison of the design, installation, and maintenance treatment of safety related and non-safety related, PRA credited valves in the Reactor Building • Conservative treatment of MOVs in Turbine Building These are discussed as follows: 1. The safety related valves located in the Reactor Building are qualified for HELB conditions and are therefore considered robust in their ability to survive spray effects. 2. SSCs are assumed failed if the SSC is submerged. For the assessment of spray impacts on SSCs, the primary emphasis is on electrical equipment that could cause failures of multiple pieces of equipment. EPRI document 1019194, December 2009 “Guidelines for Performance of Internal Flooding Probabilistic Risk Assessment”, indicates the following with respect to water spray effects: • Water spray is assumed to fail electrical equipment such as

		switchgear and motor control centers (MCCs), unless protected by suitably installed shields. The evaluation should differentiate between moderate-energy piping systems (maximum operating pressure less than 275 psig) and high-energy piping systems. For Fermi 2, the switchgear and MCCs are explicitly evaluated for spray effects. This is documented in Appendix B.5 where it is stated that the EPRI guidelines for internal flooding analysis are followed which require the examination of spray effects on switchgear and MCCs. 3. The valves that are cited in the proposed finding are: • Picture 251: There are no spray sources that can affect this valve. Rupture failure of General Service Water (GSW) fails all mitigation equipment in the Turbine Building. • Picture 248: The two AOVs and SOVs are BOP hotwell makeup valves. As pointed out, these valves are assumed to fail for all floods in the Turbine Building. Therefore, no credit is attached to these for flood scenarios and no other systems are present (as shown in the picture) such that spray from another system can simultaneously fail these valves and an additional mitigation system. 4. The Fermi 2 MOV/AOV Valve Engineer examined the Internal Flooding Walkdown documentation with respect to the valves (including SOVs) identified in the pictures discussed in this finding and also regarding non-safety related, PRA-credited valves. It was noted that the valves in the pictures were very similar to valves throughout the plant, including safety related valves, in terms of spray resistance. It was also his judgment (based upon information in the response to Finding 4-21) that, since safety related and non-safety related, PRA-credited valves in the plant are similar in design, installation, and maintenance treatment, the two classes of valves would perform similarly during spray events. Based on the Fermi 2 implementation of the EPRI Internal Flood Guidelines that require a search for spray effects on MCCs and switchgear and the information presented here regarding spray effects on valves this SR meets Capability Category II..

F&O / Status / SR Capability Category	Finding	Resolution
5-7 / Addressed SR HR-G6 Not Met	Human Reliability Analysis (EF2-PRA-004) Section 3.13 was reviewed and determined not to meet the intent of this SR. Section 3.13 of the HRA notebook simply reviews Fermi 2 human performance indicators and attempts to draw a correlation to this SR. Table 5-4 does not provide a means to evaluate HFEs given the scenario context of an accident sequence. A review of Table 5-4 did not reveal that a comparison for reasonableness was made at the time of the analysis. The intent of this standard is to assess the HFEs relative to each other, i.e., for all of the HFEs that fall within a specific range, is the expected failure rate of the operators considered reasonable? For example, are all of the events that have a 1E-1 probability considered more difficult than the HFEs that have probabilities in the 1E-2 range? Similarly all of the HFE's that have probabilities on the 1E-3 range should be generally considered to have the same level of difficulties compared to the ones in the 1E-2 range.	A comparison of the Human Failure Events (HFEs) is provided to assess the reasonableness of the Human Error Probabilities (HEPs). The HEPs are ranked by their HEP value. Then a comparison is made of all of the HEPs within a single decade. This comparison shows that the HEPs are consistently assessed quantitatively with respect to each other. Finally, the HEPs from one decade are compared with HEPs of other decades to verify that they are indeed of a significantly different character such that it justifies their different quantification. This tabular comparison and the resulting insights provide an additional reasonableness check as requested by the PRA Peer Review Team. As reasonableness checks for HFE failure rates have been completed and are documented in the HRA notebook, this SR meets Capability Category II.

B. Appendix B – PRA Model Changes for Containment Overpressure

The sensitivity case described in Section 5.4.3 required modeling modifications to quantify the sensitivity case. The modeling modifications implemented the framework described in Section 5.4.3 by implementing the following modifications:

1. Add a basic event representing the failure probability for containment leakage for Class 3b (PHPHECCS-NPSH-ILRT). This basic event has a baseline failure probability of 2.3E-3 (derived from the values inserted into Eqn. 5-3 in Section 5.0 for $Prob_{Class\ 3b}$); this value corresponds to the baseline probability for the ILRT frequency of 3 per 10 years. The probability for the 1 per 15 year case is five times the 3 per 10 year case (or 1.15E-2/yr).
2. Gate GHE1FVNTCTRL was changed from EQU to OR to include basic event PHPHECCS-NPSH-ILRT. This gate represents the inability to maintain NPSH due to containment leakage or the inability of the operators to control containment pressure within the pressure band necessary to maintain overpressure during containment venting ("over-venting").
3. Gate NO-LPI-ILRT was created to add the basic event PHPHECCS-NPSH-ILRT to the model logic where the gate NO-LPI appeared; gate NO-LPI-ILRT is an OR gate with inputs NO-LPI and PHPHECCS-NPSH-ILRT. This logic fails low pressure ECCS injection (LPCI and Core Spray) in cases where containment venting is not successful followed by failure of the containment.
4. Gates VNT-RB-EQUIP-1 and VNT-RB-EQUIP-2 were created to include the phenomenological probability for event PHPHECCS-NPSH-ILRT as a pre-condition for the evaluation of basic event CPFFRBLDFAILDUCTL1 (COND. PROB. THAT ADVERSE ENVIRONMENT FAILS EQUIPMENT IN RB BASEMENT – LEVEL 1 PRA); that is, if there is a leak in containment due to a lack of ILRT testing, there is a 50% probability (based upon the basic event probability for CPFFRBLDFAILDUCTL1) that the ECCS equipment in the reactor building basement will fail in Level 1 sequences..
5. Gates VNT-RB-EQUIP-NL-1 and VNT-RB-EQUIP-NL-2 were created for the same reasons as gates VNT-RB-EQUIP-1 and VNT-RB-EQUIP-2; these "NL" gates apply to sequences where offsite power has been recovered.
6. Gates VNT-RB-EQUIP-MU3-1 and VNT-RB-EQUIP-MU3-2 were created to include the phenomenological probability for event PHPHECCS-NPSH-ILRT as a pre-condition for the evaluation of basic event CPFFRBLDFAILDUCTL2 (COND. PROB. THAT ADVERSE ENVIRONMENT FAILS EQUIPMENT IN RB BASEMENT – LEVEL 2 PRA); that is, if there is a leak in containment due to a lack of ILRT testing, there is a 100% probability (based upon the basic event probability for CPFFRBLDFAILDUCTL2) that the ECCS equipment in the reactor building basement will fail in Level 2 sequences.
7. Gates VNT-RB-EQUIP-MU3-NL-1 and VNT-RB-EQUIP-MU3-NL-2 were created for the same reasons as gates VNT-RB-EQUIP-MU3-1 and VNT-RB-EQUIP-MU3-2; these "NL" gates apply to sequences where offsite power has been recovered.

Table B-1 below summarizes the differences between the baseline model and the model for the ILRT containment overpressure sensitivity.

Table B-1 – Differences between ILRT Sensitivity Model an Model of Record

T1[(a)] has 8071 gates
T2[(b)] has 8062 gates
In T1, GHE1FVNTCTRL inputs are HE1FVNTCTRL,PHPHECCS-NPSH-ILRT
In T2, GHE1FVNTCTRL inputs are HE1FVNTCTRL
In T1, VNT-RB-EQUIP inputs are CPFFRBLDFAILDUCTL1,VNT-RB-EQUIP-1
In T2, VNT-RB-EQUIP inputs are CPFFRBLDFAILDUCTL1,TC12,VNT-RB
In T1, QUV-PRES-L inputs are NO-LPI-ILRT,ZZ-EXT-RHRSW,ZZ-FIRE-RPVINJ
In T2, QUV-PRES-L inputs are NO-LPI,ZZ-EXT-RHRSW,ZZ-FIRE-RPVINJ
In T1, VNT-RB-EQUIP-NL inputs are CPFFRBLDFAILDUCTL1,VNT-RB-EQUIP-NL-1
In T2, VNT-RB-EQUIP-NL inputs are CPFFRBLDFAILDUCTL1,TC12-NL,VNT-RB-NL
NO-LPI is not the same type
In T1, T-CS-LPI-RUPT inputs are NO-LPI-ILRT,PHPHRPVI-LG-BRK
In T2, T-CS-LPI-RUPT inputs are NO-LPI,PHPHRPVI-LG-BRK
In T1, VNT-RB-EQUIP-MU3 inputs are CPFFRBLDFAILDUCTL2,VNT-RB-EQUIP-MU3-1
In T2, VNT-RB-EQUIP-MU3 inputs are CPFFRBLDFAILDUCTL2,TC12,VNT-RB-MU3
In T1, VNT-RB-EQUIP-MU3-NL inputs are CPFFRBLDFAILDUCTL2,VNT-RB-EQUIP-MU3-NL-1
In T2, VNT-RB-EQUIP-MU3-NL inputs are CPFFRBLDFAILDUCTL2,TC12-NL,VNT-RB-MU3
NO-LPI-ILRT is not in T2 (name may have been changed)
VNT-RB-EQUIP-1 is not in T2 (name may have been changed)
VNT-RB-EQUIP-2 is not in T2 (name may have been changed)
VNT-RB-EQUIP-NL-1 is not in T2 (name may have been changed)
VNT-RB-EQUIP-NL-2 is not in T2 (name may have been changed)
VNT-RB-EQUIP-MU3-NL-1 is not in T2 (name may have been changed)
VNT-RB-EQUIP-MU3-NL-2 is not in T2 (name may have been changed)
VNT-RB-EQUIP-MU3-1 is not in T2 (name may have been changed)
VNT-RB-EQUIP-MU3-2 is not in T2 (name may have been changed)

Notes:

(a) T1 refers to the revised sensitivity model.

(b) T2 refers to the baseline model of record (FermiV10).

C. Appendix C – Detailed Development of Corrosion Sensitivity Studies

The tables below provide the underlying detail associated with the additional sensitivities shown in Table 5-10. The calculations to produce these tables are performed in a manner described in Section 5.4.1 and Reference 26.

Table C-1 – Liner Corrosion Sensitivities (Base Case – Case 1)

Step	Description	Containment (Excluding Basemat)	Containment Basemat
1	Historical Steel Liner Flaw Likelihood / Failure Data	5.19E-03	1.30E-03
2	Age-Adjusted Steel Liner Flaw Likelihood (Year 1)	2.05E-03	5.13E-04
	Age-Adjusted Steel Liner Flaw Likelihood (Avg Years 5-10)	5.19E-03	1.30E-03
	Age-Adjusted Steel Liner Flaw Likelihood (Year 15)	1.43E-02	3.59E-03
	Age-Adjusted Steel Liner Flaw Likelihood (Avg 15 Years)	6.45E-03	1.61E-03
3	Flaw Likelihood (3 Years)	0.71%	0.18%
	Flaw Likelihood (10 Years)	4.14%	1.04%
	Flaw Likelihood (15 Years)	9.68%	2.42%
4	Likelihood of Breach in Containment Given Steel Liner Flaw	1.00%	0.10%
5	Visual Inspection Detection Failure Likelihood	10.00%	100.00%
6	Likelihood of Non-Detected Containment Leakage (3 yr)	7.12E-06	1.78E-06
	Likelihood of Non-Detected Containment Leakage (10 yr)	4.14E-05	1.04E-05
	Likelihood of Non-Detected Containment Leakage (15 yr)	9.68E-05	2.42E-05

Table C-2 – Liner Corrosion Sensitivities (Case 2 – Doubles Every 2 Years)

Step	Description	Containment (Excluding Basemat)	Containment Basemat
1	Historical Steel Liner Flaw Likelihood / Failure Data	5.19E-03	1.30E-03
2	Age-Adjusted Steel Liner Flaw Likelihood (Year 1)	4.62E-04	1.15E-04
	Age-Adjusted Steel Liner Flaw Likelihood (Avg Years 5-10)	5.19E-03	1.30E-03
	Age-Adjusted Steel Liner Flaw Likelihood (Year 15)	5.91E-02	1.48E-02
	Age-Adjusted Steel Liner Flaw Likelihood (Avg 15 Years)	1.34E-02	3.34E-03
3	Flaw Likelihood (3 Years)	0.20%	0.05%
	Flaw Likelihood (10 Years)	3.46%	0.86%
	Flaw Likelihood (15 Years)	20.07%	5.02%
4	Likelihood of Breach in Containment Given Steel Liner Flaw	1.00%	0.10%
5	Visual Inspection Detection Failure Likelihood	10.00%	100.00%
6	Likelihood of Non-Detected Containment Leakage (3 yr)	2.04E-06	5.10E-07
	Likelihood of Non-Detected Containment Leakage (10 yr)	3.46E-05	8.64E-06
	Likelihood of Non-Detected Containment Leakage (15 yr)	2.01E-04	5.02E-05

Table C-3 – Liner Corrosion Sensitivities (Doubles Every 10 Years)

Step	Description	Containment (Excluding Basemat)	Containment Basemat
1	Historical Steel Liner Flaw Likelihood / Failure Data	5.19E-03	1.30E-03
2	Age-Adjusted Steel Liner Flaw Likelihood (Year 1)	3.29E-03	8.21E-04
	Age-Adjusted Steel Liner Flaw Likelihood (Avg Years 5-10)	5.19E-03	1.30E-03
	Age-Adjusted Steel Liner Flaw Likelihood (Year 15)	8.70E-03	2.17E-03
	Age-Adjusted Steel Liner Flaw Likelihood (Avg 15 Years)	5.59E-03	1.40E-03
3	Flaw Likelihood (3 Years)	1.06%	0.26%
	Flaw Likelihood (10 Years)	4.58%	1.15%
	Flaw Likelihood (15 Years)	8.38%	2.10%
4	Likelihood of Breach in Containment Given Steel Liner Flaw	1.00%	0.10%
5	Visual Inspection Detection Failure Likelihood	10.00%	100.00%
6	Likelihood of Non-Detected Containment Leakage (3 yr)	1.06E-05	2.65E-06
	Likelihood of Non-Detected Containment Leakage (10 yr)	4.58E-05	1.15E-05
	Likelihood of Non-Detected Containment Leakage (15 yr)	8.38E-05	2.10E-05

Table C-4 – Liner Corrosion Sensitivities (Case 4 – 15% Detection)

Step	Description	Containment (Excluding Basemat)	Containment Basemat
1	Historical Steel Liner Flaw Likelihood / Failure Data	5.19E-03	1.30E-03
2	Age-Adjusted Steel Liner Flaw Likelihood (Year 1)	2.05E-03	5.13E-04
	Age-Adjusted Steel Liner Flaw Likelihood (Avg Years 5-10)	5.19E-03	1.30E-03
	Age-Adjusted Steel Liner Flaw Likelihood (Year 15)	1.43E-02	3.59E-03
	Age-Adjusted Steel Liner Flaw Likelihood (Avg 15 Years)	6.45E-03	1.61E-03
3	Flaw Likelihood (3 Years)	0.71%	0.18%
	Flaw Likelihood (10 Years)	4.14%	1.04%
	Flaw Likelihood (15 Years)	9.68%	2.42%
4	Likelihood of Breach in Containment Given Steel Liner Flaw	1.00%	0.10%
5	Visual Inspection Detection Failure Likelihood	15.00%	100.00%
6	Likelihood of Non-Detected Containment Leakage (3 yr)	1.07E-05	1.78E-06
	Likelihood of Non-Detected Containment Leakage (10 yr)	6.22E-05	1.04E-05
	Likelihood of Non-Detected Containment Leakage (15 yr)	1.45E-04	2.42E-05

Table C-5 – Liner Corrosion Sensitivities (Case 5 – 5% Detection)

Step	Description	Containment (Excluding Basemat)	Containment Basemat
1	Historical Steel Liner Flaw Likelihood / Failure Data	5.19E-03	1.30E-03
2	Age-Adjusted Steel Liner Flaw Likelihood (Year 1)	2.05E-03	5.13E-04
	Age-Adjusted Steel Liner Flaw Likelihood (Avg Years 5-10)	5.19E-03	1.30E-03
	Age-Adjusted Steel Liner Flaw Likelihood (Year 15)	1.43E-02	3.59E-03
	Age-Adjusted Steel Liner Flaw Likelihood (Avg 15 Years)	6.45E-03	1.61E-03
3	Flaw Likelihood (3 Years)	0.71%	0.18%
	Flaw Likelihood (10 Years)	4.14%	1.04%
	Flaw Likelihood (15 Years)	9.68%	2.42%
4	Likelihood of Breach in Containment Given Steel Liner Flaw	1.00%	0.10%
5	Visual Inspection Detection Failure Likelihood	5.00%	100.00%
6	Likelihood of Non-Detected Containment Leakage (3 yr)	3.56E-06	1.78E-06
	Likelihood of Non-Detected Containment Leakage (10 yr)	2.07E-05	1.04E-05
	Likelihood of Non-Detected Containment Leakage (15 yr)	4.84E-05	2.42E-05

Table C-6 – Liner Corrosion Sensitivities (Case 6 – Breach / 10% Cylinder / 1% Basement)

Step	Description	Containment (Excluding Basemat)	Containment Basemat
1	Historical Steel Liner Flaw Likelihood / Failure Data	5.19E-03	1.30E-03
2	Age-Adjusted Steel Liner Flaw Likelihood (Year 1)	2.05E-03	5.13E-04
	Age-Adjusted Steel Liner Flaw Likelihood (Avg Years 5-10)	5.19E-03	1.30E-03
	Age-Adjusted Steel Liner Flaw Likelihood (Year 15)	1.43E-02	3.59E-03
	Age-Adjusted Steel Liner Flaw Likelihood (Avg 15 Years)	6.45E-03	1.61E-03
3	Flaw Likelihood (3 Years)	0.71%	0.18%
	Flaw Likelihood (10 Years)	4.14%	1.04%
	Flaw Likelihood (15 Years)	9.68%	2.42%
4	Likelihood of Breach in Containment Given Steel Liner Flaw	10.00%	1.00%
5	Visual Inspection Detection Failure Likelihood	10.00%	100.00%
6	Likelihood of Non-Detected Containment Leakage (3 yr)	7.12E-05	1.78E-05
	Likelihood of Non-Detected Containment Leakage (10 yr)	4.14E-04	1.04E-04
	Likelihood of Non-Detected Containment Leakage (15 yr)	9.68E-04	2.42E-04

Table C-7 – Liner Corrosion Sensitivities (Case 7 - Breach / 0.1% Cyl / 0.01% Basement)

Step	Description	Containment (Excluding Basemat)	Containment Basemat
1	Historical Steel Liner Flaw Likelihood / Failure Data	5.19E-03	1.30E-03
2	Age-Adjusted Steel Liner Flaw Likelihood (Year 1)	2.05E-03	5.13E-04
	Age-Adjusted Steel Liner Flaw Likelihood (Avg Years 5-10)	5.19E-03	1.30E-03
	Age-Adjusted Steel Liner Flaw Likelihood (Year 15)	1.43E-02	3.59E-03
	Age-Adjusted Steel Liner Flaw Likelihood (Avg 15 Years)	6.45E-03	1.61E-03
3	Flaw Likelihood (3 Years)	0.71%	0.18%
	Flaw Likelihood (10 Years)	4.14%	1.04%
	Flaw Likelihood (15 Years)	9.68%	2.42%
4	Likelihood of Breach in Containment Given Steel Liner Flaw	0.10%	0.01%
5	Visual Inspection Detection Failure Likelihood	10.00%	100.00%
6	Likelihood of Non-Detected Containment Leakage (3 yr)	7.12E-07	1.78E-07
	Likelihood of Non-Detected Containment Leakage (10 yr)	4.14E-06	1.04E-06
	Likelihood of Non-Detected Containment Leakage (15 yr)	9.68E-06	2.42E-06

Table C-8 – Liner Corrosion Sensitivities (Case 8 – Lower Bound)

Step	Description	Containment (Excluding Basemat)	Containment Basemat
1	Historical Steel Liner Flaw Likelihood / Failure Data	5.19E-03	1.30E-03
2	Age-Adjusted Steel Liner Flaw Likelihood (Year 1)	3.29E-03	8.21E-04
	Age-Adjusted Steel Liner Flaw Likelihood (Avg Years 5-10)	5.19E-03	1.30E-03
	Age-Adjusted Steel Liner Flaw Likelihood (Year 15)	8.70E-03	2.17E-03
	Age-Adjusted Steel Liner Flaw Likelihood (Avg 15 Years)	5.59E-03	1.40E-03
3	Flaw Likelihood (3 Years)	1.06%	0.26%
	Flaw Likelihood (10 Years)	4.58%	1.15%
	Flaw Likelihood (15 Years)	8.38%	2.10%
4	Likelihood of Breach in Containment Given Steel Liner Flaw	0.10%	0.01%
5	Visual Inspection Detection Failure Likelihood	5.00%	100.00%
6	Likelihood of Non-Detected Containment Leakage (3 yr)	5.29E-07	2.65E-07
	Likelihood of Non-Detected Containment Leakage (10 yr)	2.29E-06	1.15E-06
	Likelihood of Non-Detected Containment Leakage (15 yr)	4.19E-06	2.10E-06

Table C-9 – Liner Corrosion Sensitivities (Case 9 - Upper Bound)

Step	Description	Containment (Excluding Basemat)	Containment Basemat
1	Historical Steel Liner Flaw Likelihood / Failure Data	5.19E-03	1.30E-03
2	Age-Adjusted Steel Liner Flaw Likelihood (Year 1)	4.62E-04	1.15E-04
	Age-Adjusted Steel Liner Flaw Likelihood (Avg Years 5-10)	5.19E-03	1.30E-03
	Age-Adjusted Steel Liner Flaw Likelihood (Year 15)	5.91E-02	1.48E-02
	Age-Adjusted Steel Liner Flaw Likelihood (Avg 15 Years)	1.34E-02	3.34E-03
3	Flaw Likelihood (3 Years)	0.20%	0.05%
	Flaw Likelihood (10 Years)	3.46%	0.86%
	Flaw Likelihood (15 Years)	20.07%	5.02%
4	Likelihood of Breach in Containment Given Steel Liner Flaw	10.00%	1.00%
5	Visual Inspection Detection Failure Likelihood	15.00%	100.00%
6	Likelihood of Non-Detected Containment Leakage (3 yr)	3.06E-05	5.10E-06
	Likelihood of Non-Detected Containment Leakage (10 yr)	5.18E-04	8.64E-05
	Likelihood of Non-Detected Containment Leakage (15 yr)	3.01E-03	5.02E-04