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Detroit Edison

A DTE Energy Company

10CFR50.90

May 23, 2002
NRC-02-0040

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

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

Subject: Proposed License Amendment for a One-Time Deferral of the Primary Containment Integrated Leak Rate Test

Pursuant to 10 CFR 50.90, Detroit Edison hereby proposes to amend the Fermi 2 Plant Operating License, Appendix A, Technical Specifications (TS) to allow a one-time deferral of the Type A primary containment integrated leak rate test (ILRT). Specifically, TS 5.5.12, "Primary Containment Leakage Rate Testing Program," is proposed to be revised to extend the current interval for performing the containment Type A test to 15 years.

Enclosure 1 provides a description and an evaluation of the proposed change. Enclosure 2 provides an analysis of the issue of significant hazards consideration using the standards of 10 CFR 50.92. Enclosure 3 provides a marked up page of the existing TS to show the proposed change and a typed version of the affected TS page with the proposed change incorporated. Enclosure 4 provides a report on the risk assessment regarding the ILRT interval extension at Fermi 2.

Detroit Edison has reviewed the proposed change against the criteria of 10 CFR 51.22 for environmental considerations. The proposed change does not involve a significant hazards consideration, nor does it significantly change the types or significantly increase the amounts of effluents that may be released offsite. The proposed change does not significantly increase individual or cumulative occupational radiation exposures. Based on the foregoing, Detroit Edison concludes

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that the proposed change meets the criteria provided in 10 CFR 51.22 (c) (9) for a categorical exclusion from the requirements for an Environmental Impact Statement or an Environmental Assessment.

Detroit Edison requests NRC approval of this license amendment by November 22, 2002, with an implementation period of within 60 days following NRC approval. This approval date is requested to allow for adequate planning and scheduling of the upcoming ninth refueling outage, scheduled to start in March 2003.

Should you have any questions or require additional information, please contact Mr. Norman K. Peterson of my staff at (734) 586-4258.

Sincerely,

Enclosures

cc: T. J. Kim
M. A. Ring
NRC Resident Office
Regional Administrator, Region III
Supervisor, Electric Operators,
Michigan Public Service Commission

I, WILLIAM T. O'CONNOR, JR., do hereby affirm that the foregoing statements are based on facts and circumstances which are true and accurate to the best of my knowledge and belief.

WILLIAM T. O'CONNOR, JR.
Vice President - Nuclear Generation

On this 23rd day of May, 2002 before me personally appeared William T. O'Connor, Jr., being first duly sworn and says that he executed the foregoing as his free act and deed.

Karen M Reed
Notary Public

KAREN M. REED
Notary Public, Monroe County, MI
My Commission Expires 09/02/2005

NRC-02-0040
ENCLOSURE 1

FERMI 2 NRC DOCKET NO. 50-341
OPERATING LICENSE NO. NPF-43

REQUEST FOR A ONE-TIME DEFERRAL OF THE PRIMARY CONTAINMENT INTEGRATED LEAK RATE TEST

DESCRIPTION AND EVALUATION OF THE PROPOSED CHANGE

DESCRIPTION AND EVALUATION OF THE PROPOSED CHANGE

DESCRIPTION:

This proposed License Amendment requests the revision of TS 5.5.12 to allow for a one-time extension of the current primary containment Type A test interval from 10 to 15 years.

Fermi 2 Technical Specification (TS) 5.5.12, "Primary Containment Leakage Rate Testing Program," requires that a program 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. It further requires that this program be in accordance with the guidelines contained in Regulatory Guide (RG) 1.163, "Performance-Based Containment Leak-Test Program," dated September 1995, with the exception of approved exemptions to 10 CFR 50, Appendix J.

10 CFR 50, Appendix J, Option B requires a Type A test to be conducted at a periodic interval based on the historical performance of the overall containment system. Type A integrated leak rate tests (ILRT) measure the overall integrated leak rate of the containment structure; Type B tests measure leakage across containment pressure boundary seals, gaskets and expansion bellows; and Type C tests measure leakage rates of containment isolation valves.

RG 1.163 endorses, with certain exceptions, Nuclear Energy Institute (NEI) 94-01, Revision 0, "Industry Guideline for Implementing Performance-Based Option of 10 CFR Part 50, Appendix J," dated July 26, 1995. NEI 94-01 specifies an initial test interval of 48 months, but allows an extended interval of 10 years, based upon two consecutive successful tests. There is also a provision for extending the test interval an additional 15 months beyond the 10-year interval in certain circumstances.

As shown in the table below, the two most recent Type A leak rate tests conducted at Fermi 2 have been successful. The as-left leakage met the acceptance criterion of 0.75 of the maximum allowable leak rate per day (La); therefore, the current Type A leakage rate test interval is 10 years.

Test Date	Upper Confidence Level Measured Leakage	Correction for Type B and C Tests	Total Leakage	Acceptance Criteria
11/22/1989	0.285 La	0.033 La	0.318 La	0.75 La
10/28/1992	0.212 La	0.032 La	0.244 La	0.75 La

Fermi 2 utilizes a General Electric (GE) Boiling Water Reactor (BWR) Mark I primary containment structure. The containment consists of an inverted bulb-shaped steel liner vessel (drywell) and a pressure suppression chamber (torus). Eight downcomer vent pipes connecting

the drywell to the torus and extending into the water in the torus are used for venting the drywell atmosphere following a loss-of-coolant accident (LOCA). There are several penetrations used for access into the containment and for other process piping and electrical service.

The drywell 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 bottom portion of the drywell shell is totally embedded in concrete. There is a two-foot thick layer of compacted sand at 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 any condensation that may accumulate in the 2-inch gap outside the drywell steel liner. Four drain lines are used to remove any moisture in the sand cushion. A small portion of the sand has been removed from the drain line locations to enhance drainage capability from the sand cushion cavity.

In addition to the drywell head, 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 hatch covers are bolted in place and sealed with gaskets. Provisions have been made to permit leakage testing of the drywell head, CRD hatch, personnel door and equipment hatch cover seals. The two suppression pool manholes are also designed with leak testing capability. In addition to the Type A test, the integrity of the primary containment penetrations and isolation valves are verified through Type B and Type C local leak rate tests (LLRTs) as required by 10 CFR Part 50, Appendix J. These tests are performed to verify the leak-tight characteristics of the containment at the design basis accident pressure.

Deferral of the Type A containment leak rate test will result in significant savings in cost and unnecessary personnel radiation exposure during the next refueling outage without significant increase in the risk to the health and safety of the public. Cost savings are estimated at approximately $1.5 million, which include the cost of performing the test and replacement power during the critical path outage time needed to perform the test. Personnel radiation exposure reduction is estimated at about 2.0 rem.

This proposed License Amendment requests the revision of TS 5.5.12, to allow for a deviation from the Type A test interval guidelines in RG 1.163. Specifically, the proposed addition to TS 5.5.12 would allow an exception to the Type A testing frequency specified in NEI 94-01, paragraph 9.2.3, and would result in a one-time extension of the current Type A test interval from 10 to 15 years.

The NRC approved similar License Amendments for the one-time extension of containment Type A test intervals to 15 years for several nuclear facilities including Peach Bottom Atomic

Power Station, Unit 3 (ADAMS Accession No. ML012210108), Brunswick Steam Electric Plant, Unit1 (ADAMS Accession No. ML020670684) and Susquehanna Steam Electric Station, Units 1 and 2 (ADAMS Accession No. ML020280102)

EVALUATION OF THE PROPOSED CHANGES:

Containment Inspection

The ILRT, LLRTs and inservice inspection (ISI) program for the primary containment collectively ensure the leak-tightness and structural integrity of the containment. Fermi 2 is using the 1992 Edition and the 1992 Addenda of Subsection IWE of Section XI of the ASME Boiler and Pressure Vessel Code, with approved relief for certain Code requirements, for conducting the ISI of the Fermi 2 containment. Visual examinations and leak rate testing are required per TS 5.5.6 and TS surveillance SR 3.6.1.1.1. The expedited examination of containment required by 10 CFR 50.55a(g)(6)(ii)(B) was completed during the seventh refueling outage (RFO7) in the Spring of 2000. The visual inspections identified a small pit at the interface of an I-beam with the containment steel liner, minor material loss on a single tie-down eyebolt on the north equipment hatch, a small crack in the rubber seal for the outer drywell airlock door, and some localized protective coating degradation. All identified conditions have been repaired or dispositioned.

Fermi 2 has a Mark I primary containment. The configuration of a Mark I containment has three primary areas susceptible to degradation. These are the (1) 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. These three areas are addressed below:

- In RFO7, during the first containment inspection period, the entire 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. Recent re-inspections during the eighth refueling outage in November 2001 confirmed no degradation of the liner or seal area.

- In response to concerns regarding the potential degradation of the uninspectable areas of the drywell liner, work was initiated to clean the drain lines in the sand cushion area and to perform a video probe inspection for any moisture trapped in the sand cushion region. The cleaning of the four drain lines and inspection for moisture was completed during the fourth refueling outage in 1994. The inspection concluded that the drains were dry and the sand cushion area was free of any signs of moisture.

Since the initial inspection in 1994, all four drains 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. No signs of moisture have been found since the initial inspection.

- Inspection of the torus interior is performed every two refueling outages (about every three years). The inspection is performed by personnel certified in conducting visual inspections of the containment pressure boundary surfaces and the applied protective coating. The last inspection, performed in October 2001, included all interior surfaces of the torus, both above and below water. Several areas of minor coating degradation were identified and repaired. No areas of material loss of the primary containment shell were found. This ongoing inspection and repair program ensures the continued integrity of the torus.

Based on the above, no containment augmented inspections as specified in article IWE-1240 of Section XI of the ASME Code, have been identified at Fermi 2.

For the leak tightness of seals, gaskets and bolted connections, Detroit Edison 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, at least once each containment inspection interval. Bolted connections are tested per Appendix J and are subject to a general visual inspection once each containment inspection period. Detroit Edison 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.

NRC Information Notice (IN) 92-20, "Inadequate Local Leak Rate Testing," discussed inadequate Type B local leak rate testing of two-ply stainless steel bellows. Detroit Edison determined, based on a review of the purchase specifications and discussion with the manufacturers, 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.

During power operation, the primary containment atmosphere is inerted with nitrogen to ensure that no external sources of oxygen are introduced into containment. The containment atmosphere control system is used to initially purge the primary containment and to provide a supply of makeup nitrogen to maintain primary containment oxygen concentration within TS limits. A drywell makeup station senses the pressure of both the primary containment and the secondary containment and maintains a positive pressure in the primary containment. The primary containment pressure is also periodically monitored in the control room. The continuous pressurization and monitoring of the primary containment pressure provides a detection mechanism for any gross containment leakage.

Risk Impact Assessment

A detailed performance-based risk assessment for Fermi 2 was performed to support this License Amendment request. The assessment considered the guidelines of NEI 94-01, the methodology used in Electric Power Research Institute's (EPRI) TR-104285 report titled: "Risk Impact Assessment of Revised Containment Leak Rate Testing," dated August 1994, and RG 1.174, "An Approach for Using Probabilistic Risk Assessment in Risk-Informed Decisions on Plant-Specific Changes to the Licensing Basis," dated July 1998. The assessment also followed the guidance and additional information distributed by NEI in November 2001 to their Administrative Points of Contact regarding risk assessment of one-time extensions of containment ILRT intervals.

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

The EPRI study used an analytical approach similar to that presented in NUREG-1493 for evaluating the incremental risk associated with increasing the interval for Type A tests. The EPRI study estimated that relaxing the test frequency from 3 in 10 years to 1 in 10 years, will increase the average time during which a leak, detectable only by a Type A test, goes undetected, from 18 to 60 months. Industry leakage rate data gathered from 1987 to 1993 indicate that 97 percent of leaks are identified during LLRTs, and only about 3 percent are detected during Type A tests. Therefore, the Type A test frequency relaxation corresponds to about a 10 percent increase in the overall probability of leakage. The EPRI study confirmed the NUREG-1493 conclusion that a reduction in the frequency of Type A tests from 3 per 10 years to 1 per 10 years leads to a 0.02 to 0.14 percent increase in risk contribution of pre-existing leakage.

Building upon the methodology of the EPRI study and following the guidance published by NEI in November 2001, Detroit Edison assessed the change in the predicted population dose rate associated with the interval extension. The assessment also evaluated the risk increase resulting from extending the ILRT interval in terms of Large Early Release Frequency (LERF), and the impact on Conditional Containment Failure Probability (CCFP). RG 1.174 provides guidance for using Probabilistic Risk Assessment (PRA) in risk-informed decisions for determining the risk impact of plant-specific changes to the licensing basis. The RG defines very small changes in the risk acceptance guidelines as increases in Core Damage Frequency (CDF) of less than 1E-06 per reactor year and increases in LERF of less than 1E-07 per reactor year. Since the Type A test does not impact CDF, the only relevant criterion is the change in LERF. RG 1.174 also encourages the use of risk analysis techniques to help ensure and demonstrate that key principles,

such as defense-in-depth philosophy, are met. Based on that, the increase in the CCFP, which helps to ensure that the defense-in-depth philosophy is maintained, was evaluated.

The Fermi 2 risk assessment resulted in the following estimates:

- Increasing the current 10-year ILRT interval to 15 years results in an insignificant increase in total population dose rate. The total dose increases from 10.1979 to 10.1981 person-rem/year.

- Increasing the 10-year ILRT interval to 15 years results in a 7.18E-9/year increase in the LERF measure. This LERF increase is categorized as a "very small" increase per RG 1.174.

- The increase in the CCFP resulting from increasing the 10-year ILRT interval to 15 years is only 0.2 percent. Since this increase is negligible from a risk perspective, the defense-in-depth philosophy is not significantly impacted and will be maintained.

Summary

In summary, the proposed TS change is acceptable because the ILRT interval extension results in an insignificant increase in the risk to the health and safety of the public and will not adversely affect the leak tight integrity of the primary containment. The LLRTs and the containment inspection program will continue to examine and monitor potential age-related and environmental degradation of the pressure retaining components of the Fermi 2 primary containment.

NRC-02-0040
ENCLOSURE 2

FERMI 2 NRC DOCKET NO. 50-341
OPERATING LICENSE NO. NPF-43

REQUEST FOR A ONE-TIME DEFERRAL OF THE PRIMARY CONTAINMENT INTEGRATED LEAK RATE TEST

10CFR50.92 SIGNIFICANT HAZARDS CONSIDERATION

10CFR50.92 SIGNIFICANT HAZARDS CONSIDERATION

In accordance with 10CFR50.92, Detroit Edison has made a determination that the proposed amendment involves no significant hazards consideration. The proposed one-time extension of the primary containment Integrated Leak Rate Test (ILRT) interval to 15 years does not involve a significant hazards consideration for the following reasons:

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

   The proposed License Amendment involves a one-time extension of the testing frequency for the primary containment 10 CFR 50, Appendix J, Type A test. The current 10-year test interval would be extended on a one-time basis to no longer than 15 years. The proposed Technical Specification (TS) change does not involve a physical plant change 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 an uncontrolled release of radioactivity to the environment resulting from postulated design basis accidents. As such, the primary containment and the testing requirements do not affect accident initiation; therefore, the proposed TS change does not involve a significant increase in the probability of an accident previously evaluated.

   Type B and C containment local leak rate testing will continue to be performed at the frequency required by the TS. As documented in NUREG-1493, "Performance-Based Containment Leakage Test Program," industry experience has shown that Type B and C tests have identified about 97 percent of containment leakage paths, and only about 3 percent have been detected by a Type A test. NUREG-1493 also concluded, in part, that reducing the frequency of Type A containment leakage rate test to once per 20 years would result in an imperceptible increase in risk. The Fermi 2 risk-based assessment of the proposed extension supports this conclusion. The design and construction of the primary containment, combined with the containment inspection program in accordance with the American Society of Mechanical Engineers (ASME) Code, Section XI, and the Maintenance Rule program per 10 CFR 50.65 requirements, provide a high degree of confidence that the containment will not degrade in a manner that is detectable only by Type A testing. Additionally, the inherent feature of Boiling Water Reactor containments which provides on-line containment monitoring capability, allows for early detection of gross containment leakage during power operation.

   Based on the above, the proposed change does not significantly increase the probability or consequences of any accident previously evaluated.

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

   The primary containment is designed to contain energy and fission products during and following design basis accidents. The containment and testing requirements, invoked to periodically demonstrate the integrity of the containment, ensure the plant's ability to mitigate the consequences of an accident; however, the containment and testing do not involve accident initiation. In addition, the proposed change to the Type A test frequency does not involve a physical change to the facility. The change does not affect the operation of the plant such that a new failure mode involving the possibility of a new or different kind of accident is created. Therefore, the proposed change does not create the potential for a new or different kind of accident from any accident previously evaluated.

3. The change does not involve a significant reduction in the margin of safety.

   The NUREG-1493 generic study on the effects of extending containment leakage testing found that reducing the Type A test frequency to once per 20 years resulted in an imperceptible increase in risk to the public. The NUREG study concluded that Type B and C testing detect most potential containment leakage. The extension of Type A test interval to 15 years has a minimal effect on leakage detection capability. The TS allowed leakage limit is not impacted by this change, and the frequency of local Type B and C testing will not be altered as a result of this extension. Additionally, the containment inspection program provides a high degree of assurance that the containment will not degrade in a manner only detectable by Type A testing. On-line containment monitoring provides additional assurance for detecting gross containment leakage during power operation. The combination of all these factors ensures that the safety margin will be maintained. Therefore, the proposed changes will not result in a significant reduction in the margin of safety.

Based on the above, Detroit Edison has determined that the proposed amendment does not involve a significant hazards consideration.

NRC-02-0040
ENCLOSURE 3

FERMI 2 NRC DOCKET NO. 50-341
OPERATING LICENSE NO. NPF-43

REQUEST FOR A ONE-TIME DEFERRAL OF THE PRIMARY CONTAINMENT INTEGRATED LEAK RATE TEST

Attached is a marked-up page of the existing TS indicating the proposed change (Part 1) and a typed version incorporating the proposed change (Part 2)

NRC-02-0040
ENCLOSURE 3
PART 1

FERMI 2 NRC DOCKET NO. 50-341
OPERATING LICENSE NO. NPF-43

PROPOSED TS MARKED UP PAGE

INCLUDED PAGE:

5.0-18

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, (e) ← INSERT A

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 $\leq$ 5 scf per hour when the gap between the door seals is pressurized to $\geq P_a$.

(continued)

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, 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.

NRC-02-0040
ENCLOSURE 3
PART 2

FERMI 2 NRC DOCKET NO. 50-341
OPERATING LICENSE NO. NPF-43

PROPOSED TS REVISED PAGE

INCLUDED PAGE:

5.0-18

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":

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 $\leq$ 5 scf per hour when the gap between the door seals is pressurized to $\geq P_a$.

(continued)

NRC-02-0040
ENCLOSURE 4

FERMI 2 NRC DOCKET NO. 50-341
OPERATING LICENSE NO. NPF-43

REQUEST FOR A ONE-TIME DEFERRAL OF THE PRIMARY CONTAINMENT INTEGRATED LEAK RATE TEST

RISK ASSESSMENT REPORT

ERIN Engineering And Research, Inc

RISK ASSESSMENT FOR ENRICO FERMI 2 TO SUPPORT ILRT (TYPE A) INTERVAL EXTENSION REQUEST

ERIN Report No. C1420102-4835

Principal Contributors

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Prepared for

Detroit Edison Company
Enrico Fermi 2 Plant

May 2002

RISK ASSESSMENT FOR ENRICO FERMI 2 TO SUPPORT ILRT (TYPE A) INTERVAL EXTENSION REQUEST

ERIN Report No. C1420102-4835

Prepared by: VINCENT ANDERSON, ERIN Date: 5/7/02

Reviewed by: LAWRENCE LEE, ERIN Date: MAY 7, 2002

Approved by: ED BURNS, ERIN Date: 7 MAY 2002

Accepted by: JORGE RAMIREZ, DTE Date: MAY 9 2002

Revisions:

Rev.	Description	Preparer/Date	Reviewer/Date	Approver/Date

TABLE OF CONTENTS

Section 1
INTRODUCTION

1.1 PURPOSE

The purpose of this analysis is to provide an assessment of the risk impact from implementing a one-time extension of the Fermi 2 containment Type A Integrated Leak Rate Test (ILRT) interval from ten years to fifteen years. The extension would allow for substantial cost savings as the ILRT could be deferred for additional scheduled refueling outages. The risk assessment follows the guidelines from NEI 94-01 [1], the methodology used in EPRI TR-104285 [2], the NEI Interim Guidance for Performing Risk Impact Assessments In Support of One-Time Extensions for Containment Integrated Leakage Rate Test Surveillance Intervals [3], NEI Additional Information for ILRT Extensions [21], and the NRC regulatory guidance on the use of Probabilistic Risk Assessment (PRA) findings and risk insights in support of a request for a change in a plant's licensing basis as outlined in Regulatory Guide 1.174 [4].

1.2 BACKGROUND

Revisions to 10CFR50, Appendix J (Option B) allow individual plants to extend the Integrated Leak Rate Test (ILRT) Type A surveillance testing requirements from three-in-ten years to at least once per 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 was less than normal containment leakage of 1.0La (allowable leakage).

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 states that NUREG-1493 [5], "Performance-Based Containment Leak Test Program," September 1995, 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.

The NRC report, Performance Based Leak Test Program, NUREG-1493 [5], 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 for a comparable BWR plant, that increasing the containment leak rate from the nominal 0.5 percent per day to 5 percent per day leads to a barely perceptible increase in total population exposure, and increasing the leak rate to 50 percent per day increases the total population exposure by less than 1 percent. Consequently, extending the ILRT interval should not lead to any substantial increase in risk. The current analysis is being performed to confirm these conclusions based on Fermi 2 specific models and available data.

Earlier ILRT frequency extension submittals have used the EPRI TR-104285 methodology to perform the risk assessment. In November 2001, NEI issued enhanced guidance (hereafter referred to as the NEI Interim Guidance) that builds on the TR-104285 methodology and intended to provide for more consistent submittals. [3,21] The NEI Interim Guidance was developed for NEI by EPRI using personnel who also developed the TR-104285 methodology. This Fermi 2 ILRT interval extension risk assessment employs the NEI Interim Guidance methodology.

It should be noted that, in addition to ILRT tests, 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 Boiler and Pressure Vessel Code (ASME 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 3 times every 10 years. 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. Type C tests are also not affected by the Type A test frequency change.

1.3 CRITERIA

Based on previously approved ILRT Extension requests, Detroit Edison uses the following risk metrics to characterize the change in risk associated with the one time ILRT extensions:

- Change in Large Early Release Frequency (LERF)
- Change in conditional containment failure probability
- Change in population dose (person-rem/yr)

Consistent with the NEI Interim Guidance, the acceptance guidelines in Regulatory Guide 1.174 [4] 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 discusses defense-in-depth and encourages the use of risk analysis techniques to show that key principles, such as the defense-in-depth philosophy, are met. Therefore, the increase in

the conditional containment failure probability, which helps to ensure that the defense-in-depth philosophy is maintained, will also be calculated.

In addition, based on the precedent of other ILRT extension requests [6,18,20], the total annual risk (person-rem/yr population dose) is examined to demonstrate the relative change in risk. (No threshold has been established for this parameter change.)

Section 2
METHODOLOGY

2.1 NEI INTERIM GUIDANCE

This analysis uses the approach outlined in the NEI Interim Guidance. [3,21] The nine steps of the methodology are:

1. Quantify the baseline (nominal three year ILRT interval) frequency per reactor year for the EPRI accident classes of interest. Note that EPRI categories 4, 5, and 6 are not affected by changes in ILRT test frequency.

2. Determine the containment leakage rates for EPRI categories 1, 3a and 3b.

3. Develop the baseline population dose (person-rem) for the applicable EPRI categories.

4. Determine the population dose rate (person-rem/year) by multiplying the dose calculated in Step (3) by the associated frequency calculated in Step (1).

5. Determine the change in probability of leakage detectable only by ILRT, and associated frequency for the new surveillance intervals of interest. Note that with increases in the ILRT surveillance interval, the size of the postulated leak path and the associated leakage rate are assumed not to change, however the probability of leakage detectable only by ILRT does increase.

6. Determine the population dose rate for the new surveillance intervals of interest.

7. Evaluate the risk impact (in terms of population dose rate and percentile change in population dose rate) for the interval extension cases.

8. Evaluate the risk impact in terms of LERF.

9. Evaluate the change in conditional containment failure probability.

The first seven steps of the methodology calculate the change in dose. The change in dose is the principal basis upon which the Type A ILRT interval extension was previously granted and is a reasonable basis for evaluating additional extensions. The eighth step in the interim methodology calculates the change in LERF and compares it to the guidelines in Regulatory Guide 1.174. Because there is no change in CDF, the change in LERF suffices as the quantitative basis for a risk informed decision per current NRC practice, namely Regulatory Guide 1.174. The ninth and final step of the interim methodology calculates the change in containment failure probability. The NRC has previously accepted similar calculations (Ref. [7], referred to as conditional containment failure probability, CCFP) as the basis for showing that the proposed change is consistent with the defense in depth philosophy. As such this last step suffices as the remaining basis for a risk informed decision per Regulatory Guide 1.174.

2.2 ASSUMPTIONS

The following ground rules are used in the analysis:

- The Fermi Level 1 and Level 2 internal events PRA models provide representative results for the analysis.
- An evaluation of the risk impact of the ILRT on shutdown risk is addressed using the generic results from EPRI TR-105189 [8].
- Radionuclide release categories are defined consistent with the EPRI TR-104285 methodology. [2]
- Per the NEI Interim Guidance, the representative containment leakage for EPRI Category 1 sequences is 1 L_a (L_a is the Technical Specification maximum allowable containment leakage rate).
- Per the NEI Interim Guidance, the representative containment leakage for EPRI Category 3a sequences is 10 L_a. [3]
- Per the NEI Interim Guidance, the representative containment leakage for EPRI Category 3b sequences is 35 L_a. [3]

- EPRI Category 3b is conservatively categorized as LERF based on the previously approved methodology [3].

- The impact on population doses from Interfacing System LOCAs 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 ISLOCA contribution to population dose is fixed, no changes on the conclusions from this analysis will result from this assumption.

- The reduction in ILRT frequency does not impact the reliability of containment isolation valves to close in response to a containment isolation signal.

2.3 INPUTS

This section summarizes the general resources available as input (Section 2.3.1) and the plant specific resources utilized (Section 2.3.2).

2.3.1 General Resources Available

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

1) NUREG/CR-3539 [10]
2) NUREG/CR-4220 [11]
3) NUREG-1273 [12]
4) NUREG/CR-4330 [13]
5) EPRI TR-105189 [8]
6) NUREG-1493 [5]
7) EPRI TR-104285 [2]

The first study is applicable because it provides one basis for the threshold that could be used in the Level 2 PSA for the size of containment leakage that is considered

significant and to be included in the model. The second study is applicable because it provides a basis of the probability for significant pre-existing containment leakage at the time of a core damage accident. The third study is applicable because it is a subsequent study to NUREG/CR-4220 that undertook a more extensive evaluation of the same database. The fourth study provides an assessment of the impact of different containment leakage rates on plant risk. The fifth study provides an assessment of the impact on shutdown risk from ILRT test interval extension. The sixth study is the NRC's cost-benefit analysis of various alternative approaches regarding extending the test intervals and increasing the allowable leakage rates for containment integrated and local leak rate tests. The seventh study is an EPRI study of the impact of extending ILRT and LLRT test intervals on at-power public risk.

NUREG/CR-3539 [10]

Oak Ridge National Laboratory (ORNL) documented a study of the impact of containment leak rates on public risk in NUREG/CR-3539. This study uses information from WASH-1400 [15] 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 [11]

NUREG/CR-4220 is a study performed by Pacific Northwest Laboratories (PNL) 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. The study calculated unavailabilities for Technical Specification leakages and "large" leakages.

NUREG/CR-4220 assessed the "large" containment leak probability to be in the range of 1E-3 to 1E-2, with 5E-3 identified as the point estimate based on 4 events in 740 reactor years and conservatively assuming a one-year duration for each event. It should be noted that all of the 4 identified large leakage events were PWR events, and the assumption of a one-year duration is not applicable to an inerted containment such as Fermi. NUREG/CR-4220 identifies inerted BWRs as having significantly improved potential for leakage detection because of the requirement to remain inerted during power operation. This calculation presented in NUREG/CR-4220 is called an "upper bound" estimate for BWRs (presumably meaning "inerted" BWR containment designs).

NUREG-1273 [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 [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 [8]

The EPRI study TR-105189 is useful to the ILRT test interval extension risk assessment because this EPRI study provides insight regarding the impact of containment testing on shutdown risk. This study performed 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 result of the study concluded that a small but measurable safety benefit is realized from extending the test intervals. For the BWR, the benefit from extending the ILRT frequency from 3 per 10 years to 1 per 10 years was calculated to be a reduction of approximately 1E-7/yr in the shutdown core damage frequency. This risk reduction is due to the following issues:

- Reduced opportunity for draindown events
- Reduced time spent in configurations with impaired mitigating systems

The study identified 7 shutdown incidents (out of 463 reviewed) that were caused by ILRT or LLRT activities. Two of the 7 incidents were RCS draindown events caused by ILRT/LLRT activities, and the other 5 were events involving loss of RHR and/or SDC due to ILRT/LLRT activities. This information was used in the EPRI study to estimate the safety benefit from reductions in testing frequencies. This represents a valuable insight into the improvement in the safety due to extending the ILRT test interval.

NUREG-1493 [5]

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

- Reduction in ILRT frequency from 3 per 10 years to 1 per 20 years results in an "imperceptible" increase in risk.
- Increasing containment leak rates several orders of magnitude over the design basis would minimally impact (0.2 – 1.0%) population risk.
- Given the insensitivity of risk to the containment leak rate and the small fraction of leak paths detected solely by Type A testing, increasing the interval between integrated leak rate tests is possible with minimal impact on public risk.

EPRI TR-104285 [2]

Extending the risk assessment impact beyond shutdown (the earlier EPRI TR-105189 study), the EPRI TR-104285 study is a quantitative evaluation of the impact of extending Integrated Leak Rate Test (ILRT) and (Local Leak Rate Test) 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 used a simplified Containment Event Tree to subdivide representative core damage sequences into eight (8) categories of containment response to a core damage accident:

1. Containment intact and isolated
2. Containment isolation failures due to support system or active failures
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 failure due to core damage accident phenomena
8. Containment bypass

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

> *"These study results show that the proposed CLRT* [containment leak rate tests] *frequency changes would have a minimal safety impact. The change in risk determined by the analyses is small in both absolute and relative terms. For example, for the PWR analyzed, the change is about 0.02 person-rem per year . . ."*

2.3.2 Plant Specific Inputs

The Fermi 2 specific information used to perform this ILRT interval extension risk assessment includes the following:

- Fermi Level 1 PRA
- Fermi Level 2 PRA
- Past Fermi ILRT results to demonstrate adequacy of the administrative and hardware issues.[1]

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(1) The two most recent Type A tests at Fermi 2 have been successful, so the current Type A test interval requirement is 10 years. [16]

Fermi Level 1 PRA

The Fermi Level 1 internal-events PRA used as input to this analysis is characteristic of the as-built, as-operated plant. The current Level 1 PRA model is developed in CAFTA and was quantified at a truncation of 1E-12/yr, resulting in a total core damage frequency (CDF) of 3.28E-6/yr. [17] Table 2-1 summarizes the Fermi Level 1 PRA frequency results by core damage accident class.

Fermi Level 2 PRA

The Fermi Level 2 internal-events PRA is used to calculate the release frequencies for the accidents evaluated in this assessment. The Level 2 PRA is also developed in CAFTA and quantified at a truncation of 1E-12/yr. Table 2-2 summarizes the pertinent Fermi Level 2 PRA results in terms of release category. [17] The total Large Early Release Frequency (LERF), which corresponds to the Fermi "H/E" release category, was calculated to be 2.49E-7/yr. The total release frequency is 2.34E-6/yr. This results in a calculated severe accident frequency with an intact containment frequency (i.e., containment leakage within Technical Specifications) of 9.46E-7/yr.

The Fermi Level 2 PRA was also used to define the source terms for the various EPRI categories. Radionuclide release fractions were obtained from thermal hydraulic calculations (using the MAAP code) performed for various accidents in support of the Fermi 2 IPE Submittal. These source terms were then used as input to the ex-plant consequence calculation, using the MACCS computer code, to calculate the population dose.

Table 2-1

SUMMARY OF FERMI LEVEL 1 PRA RESULTS [17]

Core Damage Accident Class	Description	CDF (/yr)	% of CDF
IA	Loss of Makeup at High RPV Pressure	8.89E-08	2.7
IBE	Early Station Blackout (less than 7 hours)	8.87E-08	2.7
IBL	Late Station Blackout (greater than 7 hours)	8.59E-07	26.2
ID	Loss of Makeup at Low RPV Pressure (transient Initiators)	2.01E-07	6.1
II	Loss of Decay Heat Removal	1.65E-06	50.2
IIIC	Loss of Makeup at Low RPV Pressure (large LOCA Initiators)	1.78E-08	0.5
IVA(1)	ATWS	3.44E-07	10.5
V	Containment Bypass	3.41E-08	1.0
Total		3.28E-06(2)	100.0%

NOTES:

(1) Class IC (Mitigated ATWS sequences with subsequent coolant make-up failure) results are included in Class IVA.

(2) The total CDF is calculated for internal event challenges with the exception of internal floods. Internal flooding initiated accidents are not explicitly included in the 3.28E-6/yr CDF total. The internal flooding induced CDF at Fermi is minimal (about 3%of total CDF).[22]

Table 2-2

SUMMARY OF FERMI LEVEL 2 PRA RESULTS [17]

Level 1 CDF		Fermi Level 2 PRA Release Bin Frequencies (1),(2),(3)													
Class	CDF	Intact (OK)	LL/E	LL/I	LL/L	L/E	L/I	L/L	M/E	M/I	M/L	H/E	H/I	H/L	Total
IA	8.89E-08	8.55E-08	N/A	1.06E-10	NA	3.09E-11	2.46E-11	4.45E-10	4.42E-12	3.58E-10	4.06E-11	2.44E-09	NA	NA	3.45E-09
IBE	8.87E-08	6.89E-08	N/A	NA	NA	6.55E-10	8.21E-10	NA	4.91E-10	NA	NA	1.78E-08	NA	NA	1.98E-08
IBL	8.59E-07	7.33E-07	N/A	3.73E-11	NA	NA	2.36E-08	3.82E-11	NA	1.69E-09	NA	NA	1.00E-07	NA	1.25E-07
ID	2.01E-07	1.68E-08	N/A	7.65E-12	8.05E-12	NA	1.15E-10	6.42E-10	5.71E-09	7.94E-11	4.33E-11	1.77E-07	NA	NA	1.84E-07
II	1.65E-06	4.17E-08	N/A	NA	8.40E-08	NA	NA	1.10E-06	NA	NA	8.02E-08	NA	NA	3.45E-07	1.61E-06
IIIC	1.78E-08	2.28E-10	N/A	NA	NA	NA	NA	NA	6.87E-10	NA	NA	1.68E-08	NA	NA	1.75E-08
IVA	3.44E-07	0.00E+00	N/A	NA	NA	1.98E-09	NA	NA	3.42E-07	NA	NA	3.64E-10	NA	NA	3.44E-07
V	3.41E-08	0.00E+00	N/A	NA	NA	NA	NA	NA	NA	NA	NA	3.41E-08	NA	NA	3.41E-08
Total	3.28E-06	9.46E-07	0.00E+00	1.50E-10	8.40E-08	2.66E-09	2.46E-08	1.10E-06	3.49E-07	2.12E-09	8.03E-08	2.49E-07	1.00E-07	3.45E-07	2.34E-06

NOTES TO TABLE 2-2:

(1) Level 2 quantified at a truncation limit value of 1E-12/yr.

(2) "N/A" indicates that the accident class did not contribute to a release of that specific category.

(3) Release bin nomenclature is [Release Magnitude]/[Timing of Release], where:

LL:	Low-Low	E:	Early
L:	Low	I:	Intermediate
M:	Moderate	L:	Late
H:	High		

Section 3
ANALYSIS

3.1 BASELINE ACCIDENT CATEGORY FREQUENCIES (STEP 1)

The first step of the NEI Interim Guidance is to quantify the baseline frequencies for each of the EPRI TR-104285 accident categories. This portion of the analysis is performed using the Fermi Level 1 and Level 2 PRA results. The results for each EPRI category are described below.

Frequency of EPRI Category 1

This group consists of all core damage accident sequences in which the containment is initially isolated and remains intact throughout the accident (i.e., containment leakage at or below maximum allowable Technical Specification leakage). Per NEI Interim Guidance, the frequency per year for this category is calculated by subtracting the frequencies of EPRI Categories 3a and 3b (see below) from the sum of all severe accident sequence frequencies in which the containment is initially isolated and remains intact (i.e., accidents classified as "OK" in the Fermi Level 2 PRA).

As discussed previously in Section 2.3.2, the frequency of the Fermi Level 2 PRA "OK" accident bin is 9.46E-7/yr. As described below, the frequencies of EPRI Categories 3a and 3b are 4.32E-8/yr and 4.32E-9/yr, respectively. Therefore, the frequency of EPRI Category 1 is calculated as (9.46E-7/yr) – (4.32E-8/yr + 4.32E-9yr) = 8.99E-7/yr.

Frequency of EPRI Category 2

This group consists of all core damage accident sequences in which the containment isolation system function fails due to failures-to-close of large containment isolation

valves (either due to support system failures; or random or common cause valve failures).

The frequency of this EPRI category is estimated as follows:

- Results (i.e., cutsets) of all Fermi Level 2 PRA containment isolation failure accident sequences (sequences 1A143, 1BE143, 1BL143, 1D143, and 3C43) combined.

- All basic events, except those related to support system failure or random or common cause valve failures-to-close, set to 0.00.

This process resulted in cutsets totaling 2.46E-10/yr. The cutsets surviving the 1E-12/yr quantification truncation limit include SBO scenarios in which the WW-RB vacuum breakers fail open (due to loss of pneumatic supply) and the redundant check valves randomly fail to remain closed.

Frequency of EPRI Category 3a

This group consists of all core damage accident sequences in which the containment is failed due to a pre-existing "small" leak in the containment structure or liner that would be identifiable only from an ILRT (and thus affected by ILRT testing frequency). Consistent with NEI Interim Guidance [21], the frequency per year for this category is calculated as:

Frequency 3a = [3a conditional failure probability] x [CDF - (CDF with independent LERF + CDF that cannot cause LERF)]

The 3a conditional failure probability (2.7E-2) value is the conditional probability of having a pre-existing "small" containment leak that is detectable only by ILRTs. This value is derived in Reference [3] and is based on data collected by NEI from 91 plants. This value is also assumed reflective of ILRT testing frequencies of 3 tests in 10 years.

The pre-existing leakage probability is multiplied by the residual core damage frequency (CDF) determined as the total CDF minus the CDF for those individual sequences that either may already (independently) cause a LERF or could never cause a LERF. As discussed previously in Section 2.3.2, the Fermi total core damage frequency is 3.28E-6/yr. Of this total CDF, the following core damage accidents involve either LERF directly (containment bypass) or will never result in LERF:

- Loss of Containment Heat Removal accidents (Fermi PRA Class II): 1.65E-6/yr
- Containment Bypass accidents (Fermi PRA Class V): 3.41E-8/yr

Therefore, the frequency of EPRI Category 3a is calculated as
[(3.28E-6/yr) – (1.65E-6/yr + 3.41E-8/yr)] x (2.70E-02) = 4.32E-8/yr.

Frequency of EPRI Category 3b

This group consists of all core damage accident sequences in which the containment is failed due to a pre-existing "large" leak in the containment structure or liner that would be identifiable only from an ILRT (and thus affected by ILRT testing frequency). Similar to Category 3a, the frequency per year for this category is calculated as:

Frequency 3b = [3b conditional failure probability] x [CDF - (CDF with independent LERF + CDF that cannot cause LERF)]

The 3b failure probability (2.7E-3) value is the conditional probability of having a pre-existing "large" containment leak that is detectable only by ILRTs. This value is derived in Reference [3] and is based on data collected by NEI from 91 plants. This value is also assumed reflective of ILRT testing frequencies of 3 tests in 10 years.

Therefore, similar to EPRI Category 3b, the frequency of Category 3a is calculated as [(3.28E-6/yr) – (1.65E-6/yr + 3.41E-8/yr)] x (2.70E-03) = 4.32E-9/yr.

Frequency of EPRI Category 4

This group consists of all core damage accident sequences in which the containment isolation function is failed due to a pre-existing failure-to-seal of Type B component(s) that would not be identifiable by an ILRT. Per NEI Interim Guidance, because this category of failures is only detected by Type B tests and not by the Type A ILRT, this group is not evaluated further in this analysis.

Frequency of EPRI Category 5

This group consists of all core damage accident sequences in which the containment isolation function is failed due to a pre-existing failure-to-seal of Type C component(s) that would not be identifiable by an ILRT. Per NEI Interim Guidance, because this category of failures is only detected by Type C tests and not by the Type A ILRT, this group is not evaluated further in this analysis.

Frequency of EPRI Category 6

This group consists of all core damage accident sequences in which the containment isolation function is failed due to "other" pre-existing failure modes (e.g., pathways left open or valves that did not properly seal following test or maintenance activities) that would not be identifiable by containment leak rate tests. Per NEI Interim Guidance, because this category of failures is not impacted by leak rate tests, this group is not evaluated further in this analysis.

Frequency of EPRI Category 7

This group consists of all core damage accident progression bins in which containment failure is induced by severe accident phenomena (e.g., overpressure). Per NEI Interim Guidance, the frequency per year for this category is based on the plant Level 2 PRA results.

As the Fermi Level 2 PRA appropriately categorizes containment failure accident sequences into different release bins, EPRI Category 7 is sub-divided in this analysis to reflect the spectrum of the Fermi Level 2 PRA results. The following sub-categories are defined here:

- Category 7a: severe accident induced containment failure resulting in Low-Low magnitude releases in Late time frame (Fermi "LL/L" release bin).

- Category 7b: severe accident induced containment failure resulting in Low magnitude releases in Late time frame (Fermi "L/L" release bin).

- Category 7c: severe accident induced containment failure resulting in High magnitude releases in Late time frame (Fermi "H/L" release bin).

- Category 7d: severe accident induced containment failure resulting in High magnitude releases in Early time frame (Fermi "H/E" release bin).

- Category 7e: all other severe accident induced containment failure scenarios not represented by categories 7a-7d.

The frequency of Category 7a is the total frequency of the Fermi Level 2 PRA "LL/L" release bin. Based on the Fermi Level 2 PRA results summarized earlier in Table 2-2, the frequency of Category 7a is 8.40E-8/yr.

The frequency of Category 7b is the total frequency of the Fermi Level 2 PRA "L/L" release bin. Based on the Fermi Level 2 PRA results summarized earlier in Table 2-2, the frequency of Category 7b is 1.10E-6/yr.

The frequency of Category 7c is the total frequency of the Fermi Level 2 PRA "H/L" release bin. Based on the Fermi Level 2 PRA results summarized earlier in Table 2-2, the frequency of Category 7c is 3.45E-7/yr.

The frequency of Category 7d is determined by subtracting the frequency of EPRI Categories 2 and 8 from the total frequency of the Fermi Level 2 PRA "H/E" release bin. The accident sequences represented by EPRI Categories 2 and 8 result in High/Early releases and are already addressed separately in this analysis, as such, their frequencies are removed from Category 7d. Based on the Fermi Level 2 results summarized earlier in Table 2-2, the frequency of the Fermi Level 2 PRA "H/E" release bin is 2.49E-7/yr. As described previously, the frequency of EPRI Category 2 is 2.46E-10/yr. As described below, the frequency of EPRI Category 8 is 3.41E-8/yr. Therefore, the frequency of Category 7d is calculated as (2.49E-7/yr) – (2.46E-10/yr + 3.41E-8/yr) = 2.15E-7/yr.

The frequency of Category 7e, 5.59E-7/yr, is determined by summing the frequencies of the remaining Fermi Level 2 PRA release bins:

- M/E: 3.49E-7
- H/I: 1.00E-7
- M/L: 8.03E-8
- L/I: 2.46E-8
- L/E: 2.66E-9
- M/I: 2.12E-9
- LL/I: 1.50E-10
- LL/E: 0.00

The release characteristics of Category 7e is modeled by the Moderate/Early (M/E) Fermi 2 release bin, given that is the largest contributor of the remaining bins as well as one of the most severe with respect to consequences.

Frequency of EPRI Category 8

This group consists of all core damage accident progression bins in which the accident is initiated by a containment bypass scenario (i.e., Break Outside Containment LOCA or Interfacing Systems LOCA, ISLOCA). The frequency of Category 8 is the total frequency of the Fermi Level 1 PRA containment bypass scenarios (Class V). Based on the Fermi Level 1 PRA results summarized earlier in Table 2-1, the frequency of Category 8 is 3.41E-8/yr.

Summary of Frequencies of EPRI Categories

In summary, per the NEI Interim Guidance, the accident sequence frequencies that can lead to radionuclide releases to the public have been derived for accident categories defined in EPRI TR-104285. The results are summarized in Table 3-1.

Table 3-1

BASELINE RELEASE FREQUENCY AS A FUNCTION OF EPRI CATEGORY

EPRI Category	Category Description	Frequency Estimation Methodology	Frequency (1/yr)
1	No Containment Failure: Accident sequences in which the containment remains intact and is initially isolated. Not affected by ILRT leak testing frequency.	Per NEI Interim Guidance: *[Total Fermi 2 "OK" release category frequency] – [Frequency EPRI Categories 3a and 3b]* *[9.46E-7/yr] – [4.32E-8/yr + 4.32E-9yr] = 8.99E-7/yr*	8.99E-7
2	Containment Isolation System Failure: Accident sequences in which the containment isolation system function fails due to failures-to-close of large containment isolation valves (either due to support system failures, or random or common cause failures). Not affected by ILRT leak testing frequency.	Cutsets of all Fermi 2 containment isolation failure sequences (1A143, 1BE143, 1BL143, 1D143, and 3C43) combined and all failure modes, except those related to support system failures or random and common cause valve failures-to-close, set to 0.00.	2.46E-10
3a	Small Pre-Existing Failures: Accident sequences in which the containment is failed due to a pre-existing small leak in the containment structure or liner that would be identifiable only from an ILRT (and thus affected by ILRT testing frequency).	Per NEI Interim Guidance: *[Fermi CDF for accidents not involving containment failure/bypass] x [2.7E-2]* *[(3.28E-6/yr) – (1.65E-6/yr + 3.41E-8/yr)] x [2.70E-02] = 4.32E-8/yr*	4.32E-8
3b	Large Pre-Existing Failures: Accident sequences in which the containment is failed due to a pre-existing large leak in the containment structure or liner that would be identifiable only from an ILRT (and thus affected by ILRT testing frequency).	Per NEI Interim Guidance: *[Fermi CDF for accidents not involving containment failure/bypass] x [2.7E-2]* *[(3.28E-6/yr) – (1.65E-6/yr + 3.41E-8/yr)] x [2.70E-03] = 4.32E-9/yr*	4.32E-9

Table 3-1

BASELINE RELEASE FREQUENCY AS A FUNCTION OF EPRI CATEGORY

EPRI Category	Category Description	Frequency Estimation Methodology	Frequency (1/yr)
4	Type B Failures: Accident sequences in which the containment is failed due to a pre-existing failure-to-seal of Type B components that would not be identifiable from a ILRT (and thus not affected by ILRT testing frequency).	Per NEI Interim Guidance: N/A (not affected by ILRT frequency)	N/A
5	Type C Failures: Accident sequences in which the containment is failed due to a pre-existing failure-to-seal of Type C components that would not be identifiable from a ILRT (and thus not affected by ILRT testing frequency).	Per NEI Interim Guidance: N/A (not affected by ILRT frequency)	N/A
6	Other Containment Isolation System Failure: Accident sequences in which the containment isolation system function fails due to "other" pre-existing failure modes not identifiable by leak rate tests (e.g., pathways left open or valves that did not properly seal following test or maintenance activities).	Per NEI Interim Guidance: N/A (not affected by ILRT frequency)	N/A
7a	Containment Failure Due to Accident (a): EPRI Category 7 applies to accident sequences in which the containment is failed due to the severe accident progression. Category 7a is defined in this analysis to apply to Fermi 2 PRA accidents that result in LL/L releases. Not affected by ILRT leak testing frequency.	*[Total Fermi 2 "LL/L" release category frequency]*	8.40E-8
7b	Containment Failure Due to Accident (b): EPRI Category 7 applies to accident sequences in which the containment is failed due to the severe accident progression. Category 7b is defined in this analysis to apply to Fermi 2 PRA accidents that result in L/L releases. Not affected by ILRT leak testing frequency.	*[Total Fermi 2 "L/L" release category frequency]*	1.10E-6

Table 3-1

BASELINE RELEASE FREQUENCY AS A FUNCTION OF EPRI CATEGORY

EPRI Category	Category Description	Frequency Estimation Methodology	Frequency (1/yr)
7c	Containment Failure Due to Accident (c): EPRI Category 7 applies to accident sequences in which the containment is failed due to the severe accident progression. Category 7c is defined in this analysis to apply to Fermi 2 PRA accidents that result in H/L releases. Not affected by ILRT leak testing frequency.	*[Total Fermi 2 "H/L" release category frequency]*	3.45E-7
7d	Containment Failure Due to Accident (d): EPRI Category 7 applies to accident sequences in which the containment is failed due to the severe accident progression. Category 7d is defined in this analysis to apply to Fermi 2 PRA accidents that result in H/E releases (excluding contributions from EPRI Categories 2 and 8). Not affected by ILRT leak testing frequency.	*[Total Fermi 2 "H/E" release category frequency] – [Frequency EPRI Categories 2 and 8]* *[2.49E-7/yr] – [2.46E-10/yr + 3.41E-8/yr] = 2.15E-7/yr*	2.15E-7
7e	Containment Failure Due to Accident (e): EPRI Category 7 applies to accident sequences in which the containment is failed due to the severe accident progression. Category 7e is defined in this analysis to apply to Fermi 2 PRA accidents that result in all other remaining release categories (consequences modeled in this assessment by M/E releases). Not affected by ILRT leak testing frequency.	Calculated as the sum of all other remaining Fermi 2 release categories: • M/E: 3.49E-7 • H/I: 1.00E-7 • M/L: 8.03E-8 • L/I: 2.46E-8 • L/E: 2.66E-9 • M/I: 2.12E-9 • LL/I: 1.50E-10 • LL/E: 0.00	5.59E-7
8	Containment Bypass Accidents: Accident sequences in which the containment is bypassed. Such accidents are initiated by LOCAs outside containment (i.e., Break Outside Containment LOCA, or Interfacing Systems LOCA). Not affected by ILRT leak testing frequency.	*[Total Fermi 2 Containment Bypass release frequency]*	3.41E-8
		TOTAL:	3.28E-6

3.2 CONTAINMENT LEAKAGE RATES (STEP 2)

The second step of the NEI Interim Guidance is to define the containment leakage rates for EPRI Categories 3a and 3b. As discussed earlier, EPRI Categories 3a and 3b are accidents with pre-existing containment leakage pathways ("small" and "large", respectively) that would only be identifiable from an ILRT.

The NEI Interim Guidance recommends containment leakage rates of 10La and 35La for Categories 3a and 3B, respectively. These values are consistent with previous ILRT frequency extension submittal applications. La is the plant Technical Specification maximum allowable containment leak rate; for Fermi 2 La is 0.5% of containment air weight per day (per Fermi 2 Technical Specification 5.5.12).

The NEI Interim Guidance describes these two recommended containment leakage rates as "conservative". The NEI recommended values of 10La and 35La are used as is in this analysis to characterize the containment leakage rates for Categories 3a and 3b.

By definition, the containment leakage rate for Category 1 (i.e., accidents with containment leakage at or below maximum allowable Technical Specification leakage) is 1.0La.

3.3 BASELINE POPULATION DOSE RATE ESTIMATES (STEPS 3-4)

The third and fourth steps of the NEI Interim Guidance are to estimate the baseline population dose (person-rem) for each EPRI category and to calculate the dose rate (person-rem/year) by multiplying the category frequencies by the estimated dose.

3.3.1 Population Dose Estimates (Step 3)

The NEI Interim Guidance recommends two options for calculating population dose for the EPRI categories:

- Use of NUREG-1150 dose calculations
- Use of plant-specific dose calculations

The NUREG-1150 [14] dose calculations were used in the EPRI TR-104285 study, as discussed previously in Section 2.3.1. The use of generic dose information for NUREG-1150 was recommended by NEI to make the ILRT risk assessment methodology more readily usable for plants that do not have a Level 3 PRA.

Although Fermi does not have a Level 3 PRA, this ILRT risk assessment employs Fermi-specific dose calculations calculated using the MACCS2 (MELCOR Accident Consequence Code System) consequence code. Appendix B provides the MACCS methodology, calculations, and results. The following discussion summarizes the population dose calculation and results.

Source Terms

One of the inputs required for the MACCS consequence calculations is source term information.

The radionuclide source terms for each EPRI category were determined from MAAP runs performed in support of the Fermi 2 IPE Submittal. A representative Fermi 2 MAAP run was identified for each of the EPRI categories of interest and the radionuclide release fractions to the environment from the MAAP run results were used to characterize the source terms of the EPRI categories. Also, the time of the initial release to the environment, and the release duration, were determined.

Refer to Appendix A for more detailed discussions of the source terms defined for each EPRI category.

General Emergency Declaration

Another input required for the MACCS consequence calculations is an estimate of the time at which a General Emergency will be declared, per plant procedures, for each of the EPRI categories. The declaration of a General Emergency is used in the MACCS calculations as the initiator for offsite public evacuation.

The Fermi 2 MAAP runs used to define the source terms are also used to define the time of General Emergency declaration.

The procedural guidance for declaration of a General Emergency is maintained at Fermi 2 in RERP Plan Implementing Procedure EP-101, Classification of Emergencies. [19]

Refer to Appendix A for more detailed discussions of the times for General Emergency declaration.

Table 3-2 summarizes the accident source terms, release timings and durations, and the time of General Emergency declaration for each of the accident scenarios used to model the EPRI accident categories.

MACCS Dose Calculations

The severe accident consequence analysis was performed by Tetra Tech NUS, Inc. using the MACCS2 computer code and Fermi site-specific information (building geometry, demographics, agricultural, and meteorological data). MACCS2 simulates the impact of severe accidents at nuclear power plants on the surrounding environment. The principal phenomena considered in MACCS are atmospheric transport, mitigative actions based on dose projections, dose accumulation by a number of pathways including food and water ingestion, early and latent health effects, and economic costs. For purposes of the Fermi 2 ILRT Risk Assessment, economic costs were not considered.

Table 3-3 summarizes the population dose calculations for each of the EPRI accident categories. The dose results are presented for the 50 mile radius around the Fermi site (this is consistent NUREG-1150 doses, past ILRT frequency extension submittals, and the NEI Interim Guidance).

Refer to Appendix B of this report for details of the MACCS2 consequence calculations (including sensitivity studies).

3.3.2 Baseline Population Dose Rate Estimates (Step 4)

The baseline dose rates per EPRI accident category are calculated by multiplying the dose estimates summarized in Table 3-3 by the frequencies summarized in Table 3-1.

The resulting baseline population dose rates by EPRI category are summarized in Table 3-4. As the conditional containment pre-existing leakage probabilities for EPRI Categories 3a and 3b are reflective of a 3-per-10 year ILRT frequency (refer to Section 3.1), the baseline results shown in Table 3-4 are indicative of a 3-per-10 year ILRT surveillance frequency.

Table 3-2

RADIONUCLIDE RELEASE FRACTIONS INPUT FOR MACCS CALCULATIONS

MACCS Run	EFILRT-1	EFILRT-2	EFILRT-3	EFILRT-4	EFILRT-5	EFILRT-6	EFILRT-7	EFILRT-8
EPRI Release Category	1	2	7a	7b	7c	7d	7e	8
Fermi 2 MAAP Run	n/a	IIIC-2,IIA-1Y	IIA-2Z	IIF-1X	IIA-1Y	IIIC-2	IVA-1	V-X1
Declaration of General Emergency	0.3 hr	0.3 hr	15 hr	15 hr	15 hrs	0.3 hr	0.1 hr	0.4 hr
Start of Release	1.5 hr	1.8 hr	37 hr	28.2 hr	37 hr	1.8 hr	1.9 hr	1.0 hr
End of Release	36 hr	36 hr	90 hr	52 hr	65 hr	36 hr	36 hr	7 hr
Noble Gases	5.00E-03	1.00	1.00	1.00	1.00	1.00	1.00	1.00
CsI	4.00E-07	2.51E-01	1.60E-04	4.00E-03	2.51E-01	2.09E-01	6.30E-02	2.17E-01
TeO2	0.00	7.60E-03	0.00	9.30E-04	7.60E-03	5.80E-02	9.00E-03	7.80E-01
SrO	0.00	4.10E-03	0.00	2.50E-03	4.10E-03	2.70E-02	2.90E-02	4.20E-02
MoO2	5.00E-10	6.60E-05	2.00E-09	1.90E-08	6.60E-05	5.10E-02	2.20E-04	2.10E-02
CsOH	4.00E-07	1.00E-02	4.10E-05	1.70E-03	1.00E-02	7.90E-02	1.40E-02	7.40E-01
BaO	3.00E-10	1.90E-03	4.70E-10	1.10E-03	1.90E-03	4.60E-02	1.30E-02	4.70E-02
La2O3	0.00	3.80E-04	0.00	9.20E-05	3.80E-04	3.60E-03	3.40E-03	5.80E-03
CeO2	0.00	2.70E-03	0.00	7.10E-04	2.70E-03	1.60E-02	1.30E-02	1.90E-02
Sb	2.00E-08	6.80E-02	1.90E-05	3.60E-02	6.80E-02	4.10E-01	2.20E-01	7.50E-01
Te2	0.00	5.60E-04	0.00	1.50E-03	5.60E-04	1.70E-02	3.80E-03	2.80E-03
UO2	0.00	1.20E-05	0.00	2.90E-06	1.20E-05	8.70E-05	8.80E-05	1.40E-04

Table 3-3

DOSE ESTIMATES FOR POPULATION WITHIN 50 MILES [1]

EPRI Category	Category Description	Person-Rem Within 50 miles
1	No Containment Failure	2.05E+02
2	Containment Isolation System Failure	4.09E+06
3a	Small Pre-Existing Failures [2]	2.05E+03
3b	Large Pre-Existing Failures [2]	7.18E+03
4	Type B Failures	n/a
5	Type C Failures	n/a
6	Other Containment Isolation System Failure	n/a
7a	Containment Failure Due to Severe Accident (a)	1.88E+04
7b	Containment Failure Due to Severe Accident (b)	1.25E+06
7c	Containment Failure Due to Severe Accident (c)	4.31E+06
7d	Containment Failure Due to Severe Accident (d)	1.21E+07
7e	Containment Failure Due to Severe Accident (e)	7.41E+06
8	Containment Bypass Accidents	1.75E+07

(1) Dose estimates are derived directly from Appendix B (Page B-10):

EPRI Category	MACCS2 Run
1	EFILRT-1
2	EFILRT-2
7a	EFILRT-3
7b	EFILRT-4
7c	EFILRT-5
7d	EFILRT-6
7e	EFILRT-7
8	EFILRT-8

(2) Dose estimates for 3a and 3b, per the NEI Interim Guidance, are calculated as 10xCategory 1 dose and 35xCategory 1 dose, respectively.

Table 3-4

BASELINE DOSE RATE ESTIMATES BY EPRI ACCIDENT CATEGORY FOR POPULATION WITHIN 50 MILES

EPRI Category	Category Description	Person-Rem Within 50 miles	Accident Frequency (Per Year)	Population Dose Rate (Person-Rem/Year Within 50 miles)
1	No Containment Failure	2.05E+02	8.99E-07	1.84E-04
2	Containment Isolation System Failure	4.09E+06	2.46E-10	1.01E-03
3a	Small Pre-Existing Failures	2.05E+03	4.32E-08	8.85E-05
3b	Large Pre-Existing Failures	7.18E+03	4.32E-09	3.10E-05
4	Type B Failures	n/a	n/a	n/a
5	Type C Failures	n/a	n/a	n/a
6	Other Containment Isolation System Failure	n/a	n/a	n/a
7a	Containment Failure Due to Severe Accident (a)	1.88E+04	8.40E-08	1.58E-03
7b	Containment Failure Due to Severe Accident (b)	1.25E+06	1.10E-06	1.37E+00
7c	Containment Failure Due to Severe Accident (c)	4.31E+06	3.45E-07	1.49E+00
7d	Containment Failure Due to Severe Accident (d)	1.21E+07	2.15E-7	2.60E+00
7e	Containment Failure Due to Severe Accident (e)	7.41E+06	5.59E-07	4.14E+00
8	Containment Bypass Accidents	1.75E+07	3.41E-08	5.97E-01
		TOTAL:	3.28E-06	10.1976

3.4 IMPACT OF PROPOSED ILRT INTERVAL (STEPS 5-9)

Steps 5 through 9 of the NEI Interim Guidance assess the impact on plant risk due to the new ILRT surveillance interval in the following ways:

- Determine change in probability of detectable leakage (Step 5)
- Determine population dose rate for new ILRT interval (Step 6)
- Determine change in dose rate due to new ILRT interval (Step 7)
- Determine change in LERF risk measure due to new ILRT interval (Step 8)
- Determine change in CCFP due to new ILRT interval (Step 9)

3.4.1 Change in Probability of Detectable Leakage (Step 5)

Step 5 of the NEI Interim Guidance is the calculation of the change in probability of leakage detectable only by ILRT (and associated re-calculation of the frequencies of the impacted EPRI categories). Note that with increases in the ILRT surveillance interval, the size of the postulated leak path and the associated leakage rates are assumed not to change; however, the probability of pre-existing leakage detectable only by ILRT does increase.

Per the NEI Interim Guidance, the calculation of the change in the probability of a pre-existing ILRT-detectable containment leakage is based on the relationship that relaxation of the ILRT interval results in increasing the average time that a pre-existing leak would exist undetected. Using the standby failure rate statistical model, the average time that a pre-existing containment leak would exist undetected is one-half the surveillance interval. For example, if the ILRT frequency is 1-per-10 years, then the average time that a leak would be undetected is 60 months (surveillance interval of 120 months divided by 2). The impact on the leakage probability due to the ILRT interval extension is then calculated by applying a multiplier determined by the ratio of the average times of undetection for the two ILRT interval cases.

As discussed earlier in Section 3.1, the conditional probability of a pre-existing ILRT-detectable containment leakage is divided into two categories. The calculated pre-existing ILRT-detectable leakage probabilities are reflective of a 3-per-10 year ILRT frequency and are as follows:

- "Small" pre-existing leakage (EPRI Category 3a): 2.70E-2
- "Large" pre-existing leakage (EPRI Category 3b): 2.70E-3

Since the latter half of the 1990's, the Fermi plant has been operating under a 1-per-10 year ILRT testing frequency consistent with the performance-based Option B of 10 CFR Part 50, Appendix J. [16] The baseline 3-per-10 year based leakage probabilities first need to be adjusted to reflect the current 1-per-10 year Fermi ILRT testing frequency, as follows:

- "Small" : 2.70E-2 x [(120 months/2) / (36 months/2)] = 9.00E-2
- "Large" : 2.70E-3 x [(120 months/2) / (36 months/2)] = 9.00E-3

Note that a nominal 36 month interval (i.e., as opposed to 40 months, 120/3) is used in the above adjustment calculation to reflect the 3-per-10 year ILRT frequency. This is consistent with operational practicalities and the NEI Interim Guidance.

Similarly, the pre-existing ILRT-detectable leakage probabilities for the 1-per-15 year ILRT frequency currently being pursued by Fermi (and the subject of this risk assessment) are calculated as follows:

- "Small" : 9.00E-2 x [(180 months/2) / (120 months/2)] = 1.35E-1
- "Large" : 9.00E-3 x [(180 months/2) / (120 months/2)] = 1.35E-2

Given the above adjusted leakage probabilities, the impacted frequencies of the EPRI categories are summarized below (refer to Table 3-1 for details regarding frequency calculations for the individual EPRI categories):

EPRI Category	EPRI Category Frequency as a Function of ILRT Interval		
	Baseline (3-per-10 year ILRT)	Current (1-per-10 year ILRT)	Proposed (1-per-15 year ILRT)
1	8.99E-07	7.88E-07	7.09E-07
3a	4.32E-08	1.44E-07	2.16E-07
3b	4.32E-09	1.44E-08	2.16E-08

Note that, per the definition of the EPRI categories, only the frequencies of Categories 1, 3a, and 3b are impacted by changes in ILRT testing frequencies.

3.4.2 Population Dose Rate for New ILRT Interval (Step 6)

The dose rates per EPRI accident category as a function of ILRT interval are summarized in Table 3-5.

3.4.3 Change in Population Dose Rate Due to New ILRT Interval (Step 7)

As can be seen from the dose rate results summarized in Table 3-5, the calculated total dose rate changes imperceptibly from the current Fermi 1-per-10 year ILRT interval to the proposed 1-per-15 year ILRT interval. The total dose increases from 10.1979 person-rem/year to 10.1981 person-rem/year (an increase of <<0.1%), respectively.

Per the NEI Interim Guidance, the change in percentage contribution to total dose rate attributable to EPRI Categories 3a and 3b is also investigated here. Using the results summarized in Table 3-5, for the current Fermi 1-per-10 year ILRT interval, the percentage contribution to total dose rate from Categories 3a and 3b is shown to be negligible:

$$[\ (2.95E\text{-}4 + 1.03E\text{-}4) / 10.1979\ ] \times 100 = 3.90E\text{-}3\%$$

For the proposed 1-per-15 year ILRT interval, the percentage contribution to total dose rate from Categories 3a and 3b increases slightly but remains negligible:

$$[\ (4.43E\text{-}4 + 1.55E\text{-}4) / 10.1981\ ] \times 100 = 5.86E\text{-}3\%$$

3.4.4 Change in LERF Due to New ILRT Interval (Step 8)

The risk increase associated with extending the ILRT interval involves the potential that a core damage event that normally would result in only a small radioactive release from an intact containment could in fact result in a larger release due to the increase in probability of failure to detect a pre-existing leak. Per the NEI Interim Guidance, only Category 3b sequences have the potential to result in large releases if a pre-existing leak were present. As such, the change in LERF (Large Early Release Frequency) is determined by the change in the frequency of Category 3b.

Category 1 accidents are not considered as potential large release pathways because the containment remains intact. Therefore, the containment leak rate is expected to be small. Similarly, Category 3a is a "small" pre-existing leak. Other accident categories such as 2, 6, 7, and 8 could result in large releases but these are not affected by the change in ILRT interval. Late releases are excluded regardless of the size of the leak because late releases are, by definition, not LERF contributors.

Table 3-5

DOSE RATE ESTIMATES AS A FUNCTION OF ILRT INTERVAL FOR POPULATION WITHIN 50 MILES

EPRI Category	Category Description	Dose Rate as a Function of ILRT Interval (Person-Rem/Yr)		
		Baseline (3-per-10 year ILRT)	Current (1-per-10 year ILRT)	Proposed (1-per-15 year ILRT)
1	No Containment Failure	1.84E-04	1.62E-04	1.45E-04
2	Containment Isolation System Failure	1.01E-03	1.01E-03	1.01E-03
3a	Small Pre-Existing Failures	8.85E-05	2.95E-04	4.43E-04
3b	Large Pre-Existing Failures	3.10E-05	1.03E-04	1.55E-04
4	Type B Failures	n/a	n/a	n/a
5	Type C Failures	n/a	n/a	n/a
6	Other Containment Isolation System Failure	n/a	n/a	n/a
7a	Containment Failure Due to Severe Accident (a)	1.58E-03	1.58E-03	1.58E-03
7b	Containment Failure Due to Severe Accident (b)	1.37E+00	1.37E+00	1.37E+00
7c	Containment Failure Due to Severe Accident (c)	1.49E+00	1.49E+00	1.49E+00
7d	Containment Failure Due to Severe Accident (d)	2.60E+00	2.60E+00	2.60E+00
7e	Containment Failure Due to Severe Accident (e)	4.14E+00	4.14E+00	4.14E+00
8	Containment Bypass Accidents	5.97E-01	5.97E-01	5.97E-01
	TOTAL:	10.1976	10.1979	10.1981

The impact on the LERF risk measure due to the proposed ILRT interval extension is calculated as follows:

delta LERF = (Frequency of EPRI Category 3b for 1-per-15 year ILRT interval) – (Frequency of EPRI Category 3b for 1-per-10 year ILRT interval)

= 2.16E-8/yr – 1.44E-8/yr

= 7.18E-9/yr

This delta LERF of 7.18E-9/yr falls into Region III, Very Small Change in Risk, of the acceptance guidelines in NRC Regulatory Guide 1.174. Therefore, increasing the ILRT interval at Fermi 2 from the currently allowed 1-per-10 years to 1-per-15 years represents a very small change in risk, and is an acceptable plant change from a risk perspective.

3.4.5 Impact on Conditional Containment Failure Probability (Step 9)

Another parameter that the NRC Guidance in Reg. Guide 1.174 states can provide input into the decision-making process is the consideration of change in the conditional containment failure probability (CCFP). The change in CCFP is indicative of the effect of the ILRT on all radionuclide releases, not just LERF. The conditional containment failure probability (CCFP) can be calculated from the risk calculations performed in this analysis. In this assessment, based on the NEI Interim Guidance, CCFP is defined such that containment failure includes all radionuclide release end states other than the intact state (EPRI Category #1) and small failures (EPRI Category #3a). The conditional part of the definition is conditional given a severe accident (i.e., core damage).

Consequently, the change in CCFP can be calculated by the difference in the Category #1 and Category #3a frequencies:

$CCFP_{\%}$ = [1 – (Intact Containment Frequency / Total CDF)] x 100%, or

= [1 – ((#1 Frequency + #3a Frequency) / CDF)] x 100%

For the 10-year interval:

$$CCFP_{10} = [1 - ((7.88E\text{-}7 + 1.44E\text{-}7) / 3.28E\text{-}6)] \times 100\%$$
$$= 71.6\%$$

And for a 15-year interval:

$$CCFP_{15} = [1 - ((7.09E\text{-}7 + 2.16E\text{-}7) / 3.28E\text{-}6)] \times 100\%$$
$$= 71.8\%$$

Therefore, the change in the conditional containment failure probability is:

$$\Delta\, CCFP_{\%} = CCFP_{15} - CCFP_{10} = 0.2 \text{ percentage points}$$

This change in CCFP of less than 1 percentage point is insignificant from a risk perspective.

Section 4
RESULTS SUMMARY

The application of the approach based on NEI Interim Guidance [3, 21], EPRI-TR-104285 [2] and previous risk assessment submittals on this subject [6, 18, 20] have led to the quantitative results summarized in this section. These results demonstrate a very small impact on risk associated with the one time extension of the ILRT test interval to 15 years.

The analysis performed examined Fermi 2 specific accident sequences in which the containment remains intact or the containment is impaired. The accidents are analyzed and the results are displayed according to the eight (8) EPRI accident categories defined in Reference [2]:

1. Containment intact and isolated
2. Containment isolation failures due to support system or active failures
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 failure due to core damage accident phenomena
8. Containment bypass

The quantitative results are summarized in Table 4-1. The key results to this risk assessment are those for the ten year interval (current Fermi condition) and the fifteen year interval (proposed change). The 3-per-10 year ILRT is a baseline starting point for this risk assessment given that the pre-existing containment leakage probabilities (estimated based on industry experience - - refer to Section 3.1) are reflective of the 3-per-10 year ILRT testing.

The following is a brief summary of some of the key aspects of the ILRT test interval extension risk analysis:

- Increasing the current 10 year ILRT interval to 15 years results in an insignificant increase in total population dose rate, from 10.1979 person-rem/year to 10.1981 person-rem/year, respectively.
- The increase in the LERF risk measure is also insignificant, a 7.18E-9/yr increase. This LERF increase is categorized as a "very small" increase per NRC Reg. Guide 1.174.
- Likewise, the conditional containment failure probability ($CCFP_{\%}$) increases insignificantly by 0.2 percentage points.

Table 4-1

QUANTITATIVE RESULTS AS A FUNCTION OF ILRT INTERVAL

EPRI Category	Dose (Person-Rem Within 50 miles)	Quantitative Results as a Function of ILRT Interval					
		Baseline (3-per-10 year ILRT)		Current (1-per-10 year ILRT)		Proposed (1-per-15 year ILRT)	
		Accident Frequency (per year)	Population Dose Rate (Person-Rem/Year Within 50 miles)	Accident Frequency (per year)	Population Dose Rate (Person-Rem/Year Within 50 miles)	Accident Frequency (per year)	Population Dose Rate (Person-Rem/Year Within 50 miles)
1	2.05E+02	8.99E-07	1.84E-04	7.88E-07	1.62E-04	7.09E-07	1.45E-04
2	4.09E+06	2.46E-10	1.01E-03	2.46E-10	1.01E-03	2.46E-10	1.01E-03
3a	2.05E+03	4.32E-08	8.85E-05	1.44E-07	2.95E-04	2.16E-07	4.43E-04
3b	7.18E+03	4.32E-09	3.10E-05	1.44E-08	1.03E-04	2.16E-08	1.55E-04
4	n/a	n/a	n/a	n/a	n/a	n/a	n/a
5	n/a	n/a	n/a	n/a	n/a	n/a	n/a
6	n/a	n/a	n/a	n/a	n/a	n/a	n/a
7a	1.88E+04	8.40E-08	1.58E-03	8.40E-08	1.58E-03	8.40E-08	1.58E-03
7b	1.25E+06	1.10E-06	1.37E+00	1.10E-06	1.37E+00	1.10E-06	1.37E+00
7c	4.31E+06	3.45E-07	1.49E+00	3.45E-07	1.49E+00	3.45E-07	1.49E+00
7d	1.21E+07	2.15E-07	2.60E+00	2.15E-07	2.60E+00	2.15E-07	2.60E+00
7e	7.41E+06	5.59E-07	4.14E+00	5.59E-07	4.14E+00	5.59E-07	4.14E+00
8	1.75E+07	3.41E-08	5.97E-01	3.41E-08	5.97E-01	3.41E-08	5.97E-01
TOTALS:		3.28E-06	10.1976	3.28E-06	10.1979	3.28E-06	10.1981
Increase in Dose Rate (1)					3E-04		2E-04
Increase in LERF (2)				1.01E-08		7.18E-09	
Increase in CCFP (%) (3)				0.3		0.2	

NOTES TO TABLE 4-1:

(1) The increase in dose rate (person-rem/year) is with respect to the results for the preceding ILRT interval, as presented in the table. For example, the increase in dose rate for the proposed 1-per-15 ILRT is calculated as: total dose rate for 1-per-15 year ILRT, 10.1981, minus total dose rate for 1-per-10 year ILRT, 10.1979, equals 2.00E-04.

(2) The increase in Large Early Release Frequency (LERF) is with respect to the results for the preceding ILRT interval, as presented in the table. As discussed in Section 3.4.4 of the report, the change in LERF is determined by the change in the accident frequency of EPRI Category 3b. For example, the increase in LERF for the proposed 1-per-15 ILRT is calculated as: 3b frequency for 1-per-15 year ILRT, 2.16E-08, minus 3b frequency for 1-per-10 year ILRT, 1.44E-08, equals 7.20E-09.

(3) As discussed in Section 3.4.5, the conditional containment failure probability (CCFP) is calculated as:

$$CCFP_{\%} = [1 - ((\text{Category \#1 Frequency} + \text{Category \#3a Frequency}) / CDF)] \times 100\%$$

Section 5
CONCLUSIONS

5.1 QUANTITATIVE CONCLUSIONS

The conclusions from the risk assessment of the one time ILRT extension can be characterized by the risk metrics used in previously approved ILRT test interval extensions. These include:

- Change in LERF
- Change in conditional containment failure probability
- Change in population dose

Based on the results from Sections 3 and 4, the main conclusion regarding the impact on plant risk associated with extending the Type A ILRT test frequency from ten years to fifteen years is:

> Reg. Guide 1.174 [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 once-per-ten years to once-per-fifteen years (using the change in the EPRI Category #3b frequency per the NEI Interim Guidance) is 7.18E-9/yr. Guidance in Reg. Guide 1.174 defines very small changes in LERF as below 10^{-7}/yr. Therefore, increasing the Fermi 2 ILRT interval from 10 to 15 years results in a very small change in risk, and is an acceptable plant change from a risk perspective.

The change in conditional containment failure probability (CCFP) is also calculated as an additional risk measure to demonstrate the impact on defense-in-depth. The ΔCCFP is found to be very small and represents a negligible change in the Fermi 2 defense-in-depth.

The change in population dose is also reported consistent with previously approved ILRT interval extension requests. The change in population dose from the current 1/10 year ILRT frequency to 1/15 year frequency is 1.96E-3%.

5.2 RISK TRADE-OFF

The performance of an ILRT introduces risk. An EPRI study of operating experience events associated with the performance of ILRTs has indicated that there are real risk impacts associated with the setup and performance of the ILRT during shutdown operation [8]. While these risks have not been quantified for Fermi 2, it is judged that there is a positive (yet unquantified) safety benefit associated with the avoidance of frequent ILRTs.

The safety benefits relate to the avoidance of plant conditions and alignments associated with the ILRT which place the plant in a less safe condition leading to events related to drain down or loss of shutdown cooling. Therefore, while the focus of this evaluation has been on the negative aspects, or increased risk, associated with the ILRT extension, there are in fact some positive safety benefits.

5.3 EXTERNAL EVENTS IMPACT

Fermi does not currently maintain external events PRA models. External event risk for Fermi was evaluated in support of the NRC one-time IPEEE (Individual Plant Examination of External Events) Program and the results were submitted to the NRC in March 1996. As such, the impact of the proposed ILRT interval extension on external event hazard risk is discussed qualitatively.

5.3.1 Overview of Fermi 2 IPEEE

Seismic Events

Consistent with NRC guidance as described in NUREG-1407, the seismic portion of the Fermi IPEEE was performed using the EPRI Seismic Margins Assessment (SMA) method. The Fermi SMA evaluation was performed for a Review Level Earthquake of 0.3g. The SMA is a deterministic evaluation process that does not calculate risk on a probabilistic basis. No core damage sequence frequencies were quantified as part of the Fermi SMA.

The Fermi plant was found to be seismically rugged. No seismic vulnerabilities were identified.

Internal Fires

Consistent with NRC guidance as described in NUREG-1407, the internal fires portion of the Fermi IPEEE was performed using the EPRI Fire-Induced Vulnerability Evaluation (FIVE) methodology. The FIVE methodology is a qualitative and quantitative screening process used to identify critical fire areas. A realistic fire-induced core damage frequency was not calculated as part of the Fermi FIVE evaluation. The critical fire areas identified using the FIVE screening process are consistent with those found at other BWR plants:

- Control Room
- Div. I Switchgear Room
- Div. II Switchgear Room
- Relay Room
- 2nd Floor of Rx. Bldg.
- Fire Compartment 11ABE (Misc. Room) in Aux. Bldg

The FIVE evaluation did not identify any unique containment failure modes (or frequency contributions) due to internal fires.

High Winds/Tornadoes

The assessment of the Fermi plant design with respect to high wind and tornado loadings determined that the plant design meets all the applicable criteria of the NRC Standard Review Plan (SRP). As such, consistent with NRC IPEEE guidance in NUREG-1407, the Fermi IPEEE evaluation concluded that core damage accidents induced by high winds or tornadoes are not significant contributors to plant risk.

External Floods

The assessment of the Fermi plant design with respect to external flooding determined that the plant design meets all the applicable criteria of the NRC Standard Review Plan (SRP). As such, consistent with NRC IPEEE guidance in NUREG-1407, the Fermi IPEEE evaluation concluded that core damage accidents induced by external flooding are not significant contributors to plant risk.

Other External Hazards

The Fermi site review and design comparison determined that all the applicable criteria of the NRC Standard Review Plan (SRP) were met. As such, consistent with NRC IPEEE guidance in NUREG-1407, the Fermi IPEEE evaluation concluded that core damage accidents induced by transportation accidents, nearby facility accidents, and other miscellaneous external hazards are not significant contributors to plant risk.

5.3.2 Qualitative Assessment of Impact on External Event Risk

Given the characteristics of this specific proposed plant change (i.e., ILRT interval extension), specific quantitative information regarding the impact on external event hazard risk measures is not a significant decision making input. The proposed ILRT interval extension impacts plant risk in a very specific and limited way, that is, it impacts a subset of accident sequences in which the probability of a pre-existing containment leak is the initial containment failure mode given a core damage accident. This impact is manifested in the plant risk profile in a similar manner for internal events and external event.

Although it is not possible at this time to perform quantitative risk assessments of external event hazards, it is judged that the impact (due to the proposed ILRT extension) on the external hazard portion of the Fermi plant risk profile is comparable to that shown previously for internal events. It is judged that if the external hazards were modeled in detail and a quantitative evaluation were performed in support of this proposed plant change, the calculated risk increase for both internal and external hazards would remain "very small".

5.4 PREVIOUS ASSESSMENTS

The NRC in NUREG-1493 [5] has previously concluded that:

- Reducing the frequency of Type A tests (ILRTs) from three per 10 years to one per 20 years was found to lead to an imperceptible increase in risk. The estimated increase in risk is very small because ILRTs identify only a few potential containment leakage paths that cannot be identified by Type B and C testing, and the leaks that have been found by Type A tests have been only marginally above existing requirements.

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

one in 20 years has not been evaluated. Beyond testing the performance of containment penetrations, ILRTs also test the integrity of the containment failure.

The findings for Fermi 2 confirm the above general findings on a plant specific basis when considering (1) Fermi 2 severe accident risk profile, (2) the Fermi 2 containment failure modes, and (3) the local population surrounding the Fermi site.

Section 6

REFERENCES

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[4] *An Approach for Using Probabilistic Risk Assessment in Risk-Informed Decisions on Plant-Specific Changes to the Licensing Basis,* Regulatory Guide 1.174, July 1998.

[5] *Performance-Based Containment Leak-Test Program,* NUREG-1493, September 1995.

[6] Letter from R.J. Barrett (Entergy) to U.S. Nuclear Regulatory Commission, IPN-01-007, dated January 18, 2001.

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[17] Letter from J V. Ramirez (Fermi2) to Linda Bugoci (Fermi 2) "*FermiV3 Results for ILRT Type A Test Frequency Extension,* " TMSA-02-0001, May 2, 2002.

[18] 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.

[19] Fermi 2 RERP Implementation Procedure, *"Classification of Emergencies"*, EP-101, Revision 26, 1/10/01.

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

[21] Letter from A. Petrangelo (NEI) to NEI Administrative Points of Cont*act, "One-Time Extension of Containment Integrated Leak Rate Test Interval – Additional Information"*, November 30, 2001.

[22] Fermi 2 Individual Plant Examination (Internal Events) dated August 1992.

Appendix A
FERMI 2 MAAP CALCULATION INFORMATION

This appendix summarizes the Fermi 2 MAAP calculations used to characterize the releases of the EPRI accident categories analyzed in this risk assessment. MAAP calculations are used to define the following inputs for the MACCS consequence calculations:

- Radionuclide release fractions
- Time of initial release
- Release duration
- Time of General Emergency declaration

A.1 MAAP RUNS

The MAAP (Modular Accident Analysis Package) code is a deterministic thermal hydraulic software package used to model severe core damage accidents. This code has been used, and continues to be used, by most U.S. nuclear utilities to provide deterministic calculations of the following information in support of PRA modeling:

- Systemic and functional success criteria
- Pressures and temperatures for various accident scenarios in the RPV, drywell, wetwell, and reactor building.
- Times for operator actions
- Times to reach these pressures and temperatures
- Source term magnitude and release timing
- Impact on source terms from containment failure locations
- General understanding of severe accidents

Detroit Edison performed numerous MAAP runs in the early 1990's in support of the Fermi 2 IPE Submittal. Fermi 2 plant specific information was used to design the Fermi MAAP models. Close to one hundred individual MAAP runs were performed.

The radionuclide release calculations have previously been identified in the Fermi 2 PRA to be consistent with the following nomenclature:

FERMI LEVEL 2 PRA RELEASE SEVERITY AND TIMING CLASSIFICATION SCHEME[1]

Release Severity		Release Timing	
Classification Category	Cs Iodide % in Release	Classification Category	Time of Initial Release[2] Relative to Time for General Emergency Declaration
High (H)	Greater than 10	Late (L)	Greater than 24 hours
Medium or Moderate (M)	1 to 10	Intermediate (I)	5 to 24 hours
Low (L)	0.1 to 1	Early (E)	Less than 5 hours
Low-low (LL)	Less than 0.1		
No iodine (OK)	0		

(1) The combinations of severity and timing classifications results in one OK release category and 12 other release categories of varying times and magnitudes.

(2) The accident initiation is used as the surrogate for the time when EALs are exceeded.

FERMI LEVEL 2 PRA RADIONUCLIDE RELEASE CATEGORIES

Time of Release	Magnitude of Release			
	H	M	L	LL
E	H/E	M/E	L/E	LL/E
I	H/I	M/I	L/I	LL/I
L	H/L	M/L	L/L	LL/L

The list of Fermi 2 IPE MAAP calculations was reviewed and appropriate cases were identified for use in this risk assessment of the ILRT interval extension. The MAAP calculations identified as appropriate are those that reflect the accident and release characteristics of the individual EPRI categories. The following Fermi 2 IPE MAAP runs are used as input to this analysis of the ILRT interval extension:

- IIIC-2
- IIA-1Y
- IIA-2Z
- IIF-1X
- IVA-1
- VX-1

MAAP run IIIC-2 is a Large LOCA accident with insufficient core cooling at t=0. The onset of core damage occurs in under an hour with RPV melt-through occurring in just under two hours. The drywell steel shell is failed due to contact with molten core debris minutes after RPV failure, resulting in a release up through the shell gap and out the blowout panels in the Refuel Floor walls (receiving little filtration by the Reactor Building in this case). The resulting release is classified as H/E. This case is used for EPRI Categories 2 and 7d.

MAAP run IIA-1Y is an MSIV closure initiated transient with initial core cooling but without adequate containment heat removal (i.e., no main condenser, RHR, or emergency containment hard-pipe vent). A large failure of the primary containment occurs at the drywell head in approximately 31.5 hours due to overpressurization. Failure of the containment is assumed to disrupt and fail all adequate core cooling. The onset of core damage occurs in about 37 hours, resulting in a release through the drywell head region and out the blowout panels in the Refuel Floor walls (receiving little filtration by the Reactor Building in this case). The resulting release is classified as H/L. This case is used for EPRI Categories 2 and 7c.

MAAP run IIA-2Z is an MSIV closure initiated transient with initial core cooling but without adequate containment heat removal (i.e., no main condenser, RHR, or emergency containment hard-pipe vent). A small primary containment failure occurs in the wetwell airspace in approximately 31.5 hours due to overpressurization. Failure of the containment is assumed to disrupt and fail all adequate core cooling. The onset of core damage occurs in about 36 hours, resulting in a release through the wetwell airspace region and out the blowout panels in the Refuel Floor walls (receiving significant filtration by the suppression pool). The resulting release is classified as LL/L. This case is used for EPRI Category 7a.

MAAP run IIF-1X is an MSIV closure initiated transient with initial core cooling but without RHR or the main condenser for adequate containment heat removal. The emergency hard-pipe containment vent is initiated in the 15-20 hour time frame per the EOPs. Loss of adequate core cooling occurs due to the venting process (e.g., flashing of suppression pool water). The onset of core damage occurs in about 25 hours; releases outside containment do not begin at this point because the containment vent is assumed reclosed. RPV melt-through occurs at about 28 hours and the drywell steel shell is failed due to contact with molten core debris minutes after RPV failure, resulting in a release up through the shell gap and out the blowout panels in the Refuel Floor walls (receiving significant filtration by the Reactor Building in this case). The resulting release is classified as L/L. This case is used for EPRI Categories 2 and 7d.

MAAP run IVA-1 is an MSIV closure initiated ATWS scenario. The unmitigated ATWS results in failing the containment in just under an hour. Failure of the containment is assumed to disrupt and fail all adequate core cooling. The onset of core damage begins at approximately 2 hours, resulting in a release that travels up the reactor building and out the blowout panels in the Refuel Floor walls (receiving little filtration by the by the reactor building in this case). The resulting release is classified as M/E. This case is used for EPRI Category 7e.

MAAP run VX-1 is a Large LOCA Outside Containment accident with insufficient core cooling at t=0. The onset of core damage occurs in under an hour with RPV melt-through occurring in just over an hour. The radionuclide release directly bypasses the primary containment and proceeds up the reactor building and out the blowout panels in the Refuel Floor walls (receiving little filtration by the Reactor Building in this case). The resulting release is classified as H/E. This case is used for EPRI Category 8.

Key accident sequence timings for each of these MAAP runs are provided in Table A-1.

A.2 SOURCE TERMS

Tables A-2 through A-9 provide summaries of the following for the EPRI categories of interest:

- Radionuclide release fractions
- Time of initial release
- Release duration

Table A-1

ACCIDENT SEQUENCE TIMINGS[1] AS A FUNCTION OF EPRI CATEGORY

EPRI Category	Category Description	Fermi 2 MAAP Case	Time to TAF	Time to 2/3 Core Height	Time to Core Damage	Time of Initial Release	Time of Gen. Emg. Declaration	Time of End of Release	Comment
1	No Containment Failure	N/A	0.2 hrs	0.3 hrs	0.8 hrs	1.5 hrs	0.3 hrs	36 hrs	General Emergency time based on Fermi EP-101, FG1
2	Containment Isolation System Failure	IIIC-2 and IIA-1Y[2]	0.2 hrs	0.3 hrs	0.8 hrs	1.8 hrs	0.3 hrs	36 hrs	General Emergency time based on Fermi EP-101, FG1
3a	Small Pre-Existing Failures	< EPRI Category 1 consequence results used; Category 1 dose multiplied by 10 for 3a >							NEI Interim Guidance
3b	Large Pre-Existing Failures	< EPRI Category 1 consequence results used; Category 1 dose multiplied by 35 for 3b >							NEI Interim Guidance
4	Type B Failures	N/A							Note (3)
5	Type C Failures	N/A							Note (3)
6	Other Containment Isolation System Failure	N/A							Note (3)

Table A-1

ACCIDENT SEQUENCE TIMINGS[1] AS A FUNCTION OF EPRI CATEGORY

EPRI Category	Category Description	Fermi 2 MAAP Case	Time to TAF	Time to 2/3 Core Height	Time to Core Damage	Time of Initial Release	Time of Gen. Emg. Declaration	Time of End of Release	Comment
7a	Containment Failure Due to Accident (a)	II-A-2-Z	34.3 hrs	35 hrs	35.9 hrs	37 hrs	15 hrs	90 hrs	General Emergency time based on Fermi EP-101, HG2[4]
7b	Containment Failure Due to Accident (b)	II-F-1-X	24 hrs	24.1 hrs	25.4 hrs	28.2 hrs	15 hrs	52 hrs	General Emergency time based on Fermi EP-101, HG2[4]
7c	Containment Failure Due to Accident (c)	II-A-1-Y	34.5 hrs	35 hrs	36.7 hrs	37 hrs	15 hrs	65 hrs	General Emergency time based on Fermi EP-101, HG2[4]
7d	Containment Failure Due to Accident (d)	III-C-2	0.2 hrs	0.3 hrs	0.8 hrs	1.8 hrs	0.3 hrs	36 hrs	General Emergency time based on Fermi EP-101, FG1
7e	Containment Failure Due to Accident (e)	IV-A-1	0.7 hrs	1.0 hrs	1.6 hrs	1.9 hrs	0.1 hrs	36 hrs	General Emergency time based on Fermi EP-101, SG2

Table A-1

ACCIDENT SEQUENCE TIMINGS[1] AS A FUNCTION OF EPRI CATEGORY

EPRI Category	Category Description	Fermi 2 MAAP Case	Time to TAF	Time to 2/3 Core Height	Time to Core Damage	Time of Initial Release	Time of Gen. Emg. Declaration	Time of End of Release	Comment
8	Containment Bypass Accidents	V-X-1	0.3 hrs	0.4 hrs	0.8 hrs	1.0 hrs	0.4 hrs	7 hrs	General Emergency time based on Fermi EP-101, FG1

NOTES TO TABLE A-1:

(1) All times listed in table referenced to t=0 (i.e., time of plant trip).

(2) Fermi 2 MAAP run IIA-1Y is used to characterize the release fractions for the Containment Isolation failure accident because this is a MAAP run in which the containment is failed prior to core damage (and the release is High). However, MAAP case IIA-1Y is a long-term loss of DHR scenario, so the accident and release timings for use in determining the time of General Emergency declaration for EPRI category 2 are approximated by the accident timings of a short-term accident modeled by Fermi MAAP run IIIC-2.

(3) Per the NEI Interim Guidance, EPRI Release Categories #4, #5, and #6 are not affected by ILRT frequency adjustments and, as such, need not be investigated.

(4) For long-term loss of containment heat removal accidents, the HG2 criterion ("Other Conditions Existing Which in the Judgment of the Emergency Director Warrant Declaration of General Emergency") of the Fermi Emergency Plan, EP-101, is used to determine the approximate time for the declaration of a General Emergency. Criterion HG2 allows Emergency Director discretion based on judgment that existing plant conditions may indicate actual or imminent substantial core degradation or melting with potential for loss of containment. The Fermi Level 2 PRA (as well as this assessment) assumes that Emergency Director discretion to declare a General Emergency will occur for loss of containment heat removal scenarios at about the time of operator initiation of containment venting (approximately 15 hours into the scenario). In such accidents, the Fermi PRA postulates that primary containment venting fails (or operates but results in failing injection systems) and that containment failure subsequently occurs (followed by core damage). Such conditions are judged here and in the Fermi PRA to warrant a reasonable assumption that the Emergency Director will invoke discretion and declare a General Emergency.

Table A-2

RADIONUCLIDE RELEASE FRACTIONS FOR EPRI CATEGORY #1[1]

EPRI Category:	#1 – No Containment Failure		
Fermi MAAP Case:	N/A[5]		
Time of Initial Release:	1.5 hrs		
End of Release:[2]	36 hrs		
	Release Fraction to Environment as a Function of Time[3][4]		
Radionuclide	**@2 hrs**	**@20 hrs**	**@36 hrs**
Noble Gases	---	---	5.00E-03
Cesium Iodide (CsI)	---	---	4.00E-07
Tellurium Oxide (TeO2)	---	---	0.0
Strontium Oxide (SrO)	---	---	0.0
Molybdenum (MoO2)	---	---	5.00E-10
Cesium Hydroxide (CsOH)	---	---	4.00E-07
Barium Oxide (BaO)	---	---	3.00E-10
Lanthanides (La2O3)	---	---	0.0
Cerium Oxide (CeO2)	---	---	0.0
Antimony (Sb)	---	---	2.00E-08
Tellurium (Te2)	---	---	0.0
Uranium Oxide (UO2)	---	---	0.0

NOTES:

(1) All times referenced to t=0 (i.e., time of plant trip).

(2) End of Release based on MAAP graphical output and is taken as either: 1) for cases in which the releases taper off well before the end of the MAAP run, the point at which approximately 95% of the cumulative CsI release occurs (estimate based on visual inspection of output graph) or 2) otherwise, the end of the MAAP run (which is typically terminated 36 hours after vessel failure).

(3) Three points in time selected based on judgment and CsI release profile of the specific Fermi 2 MAAP case. Linear behavior is assumed between points in time.

(4) "---" indicates release fraction information not available for these radionuclides. Linearly increasing release fraction from 0.0 to value at End of Release assumed.

(5) No Fermi 2 MAAP run available for this EPRI release category. Release information for this category based on review of NUREG-1150 Peach Bottom results and MAAP runs of another BWR Mark I.

Table A-3

RADIONUCLIDE RELEASE FRACTIONS FOR EPRI CATEGORY #2[1]

EPRI Category:	#2 - Containment Isolation System Failure		
Fermi MAAP Case:	IIA-1Y and IIIC-2[5]		
Time of Initial Release:	1.8 hrs		
End of Release:[2]	36 hrs		
	Release Fraction to Environment as a Function of Time[3][4]		
Radionuclide	**@2 hrs**	**@5 hrs**	**@36 hrs**
Noble Gases	0.50	1.00	1.00
Cesium Iodide (CsI)	1.00E-02	0.15	0.251
Tellurium Oxide (TeO2)	---	---	7.60E-03
Strontium Oxide (SrO)	---	---	4.10E-03
Molybdenum (MoO2)	---	---	6.60E-05
Cesium Hydroxide (CsOH)	---	---	1.00E-02
Barium Oxide (BaO)	---	---	1.90E-03
Lanthanides (La2O3)	---	---	3.80E-04
Cerium Oxide (CeO2)	---	---	2.70E-03
Antimony (Sb)	---	---	6.80E-02
Tellurium (Te2)	---	---	5.60E-04
Uranium Oxide (UO2)	---	---	1.20E-05

NOTES:

(1) All times referenced to t=0 (i.e., time of plant trip).

(2) End of Release based on MAAP graphical output and is taken as either: 1) for cases in which the releases taper off well before the end of the MAAP run, the point at which approximately 95% of the cumulative CsI release occurs (estimate based on visual inspection of output graph) or 2) otherwise, the end of the MAAP run (which is typically terminated 36 hours after vessel failure).

(3) Three points in time selected based on judgment and CsI release profile of the specific Fermi 2 MAAP case. Linear behavior is assumed between points in time.

(4) "---" indicates release fraction information not available for these radionuclides. Linearly increasing release fraction from 0.0 to value at End of Release assumed.

(5) Fermi 2 MAAP run IIA-1Y is used to characterize the release fractions for the Containment Isolation failure accident because this is a MAAP run in which the containment is failed prior to core damage (and the release is High). However, MAAP case IIA-1Y is a long-term loss of DHR scenario, so the accident and release timings for use in determining the time of General Emergency declaration are approximated by the accident timings of the short-term accident Fermi MAAP run IIIC-2.

Table A-4

RADIONUCLIDE RELEASE FRACTIONS FOR EPRI CATEGORY #7a[1]

EPRI Category:	#7a - Containment Failure Due to Accident (a)		
Fermi MAAP Case:	IIA-2Z		
Time of Initial Release:	37 hrs		
End of Release:[2]	90 hrs		
	Release Fraction to Environment as a Function of Time[3][4]		
Radionuclide	**@40 hrs**	**@70 hrs**	**@90 hrs**
Noble Gases	0.80	1.00	1.00
Cesium Iodide (CsI)	5.00E-05	7.00E-05	1.60E-04
Tellurium Oxide (TeO2)	---	---	0.00
Strontium Oxide (SrO)	---	---	0.00
Molybdenum (MoO2)	---	---	2.00E-09
Cesium Hydroxide (CsOH)	---	---	4.10E-05
Barium Oxide (BaO)	---	---	4.70E-10
Lanthanides (La2O3)	---	---	0.0
Cerium Oxide (CeO2)	---	---	0.0
Antimony (Sb)	---	---	1.90E-05
Tellurium (Te2)	---	---	0.0
Uranium Oxide (UO2)	---	---	0.0

NOTES:

(1) All times referenced to t=0 (i.e., time of plant trip).

(2) End of Release based on MAAP graphical output and is taken as either: 1) for cases in which the releases taper off well before the end of the MAAP run, the point at which approximately 95% of the cumulative CsI release occurs (estimate based on visual inspection of output graph) or 2) otherwise, the end of the MAAP run (which is typically terminated 36 hours after vessel failure).

(3) Three points in time selected based on judgment and CsI release profile of the specific Fermi 2 MAAP case. Linear behavior is assumed between points in time.

(4) "---" indicates release fraction information not available for these radionuclides. Linearly increasing release fraction from 0.0 to value at End of Release assumed.

Table A-5

RADIONUCLIDE RELEASE FRACTIONS FOR EPRI CATEGORY #7b[1]

EPRI Category:	#7b - Containment Failure Due to Accident (b)		
Fermi MAAP Case:	IIF-1X		
Time of Initial Release:	28.2 hrs		
End of Release:[2]	52 hrs		
	Release Fraction to Environment as a Function of Time[3][4]		
Radionuclide	**@30 hrs**	**@40 hrs**	**@52 hrs**
Noble Gases	1.00	1.00	1.00
Cesium Iodide (CsI)	1.50E-03	3.90E-03	4.00E-03
Tellurium Oxide (TeO2)	---	---	9.30E-04
Strontium Oxide (SrO)	---	---	2.50E-03
Molybdenum (MoO2)	---	---	1.90E-08
Cesium Hydroxide (CsOH)	---	---	1.70E-03
Barium Oxide (BaO)	---	---	1.10E-03
Lanthanides (La2O3)	---	---	9.20E-05
Cerium Oxide (CeO2)	---	---	7.10E-04
Antimony (Sb)	---	---	3.60E-02
Tellurium (Te2)	---	---	1.50E-03
Uranium Oxide (UO2)	---	---	2.90E-06

NOTES:

(1) All times referenced to t=0 (i.e., time of plant trip).

(2) End of Release based on MAAP graphical output and is taken as either: 1) for cases in which the releases taper off well before the end of the MAAP run, the point at which approximately 95% of the cumulative CsI release occurs (estimate based on visual inspection of output graph) or 2) otherwise, the end of the MAAP run (which is typically terminated 36 hours after vessel failure).

(3) Three points in time selected based on judgment and CsI release profile of the specific Fermi 2 MAAP case. Linear behavior is assumed between points in time.

(4) "---" indicates release fraction information not available for these radionuclides. Linearly increasing release fraction from 0.0 to value at End of Release assumed.

Table A-6

RADIONUCLIDE RELEASE FRACTIONS FOR EPRI CATEGORY #7c[1]

EPRI Category:	#7c - Containment Failure Due to Accident (c)		
Fermi MAAP Case:	IIA-1Y		
Time of Initial Release:	37 hrs		
End of Release:[2]	65 hrs		
	Release Fraction to Environment as a Function of Time[3][4]		
Radionuclide	**@40 hrs**	**@45 hrs**	**@65 hrs**
Noble Gases	0.50	1.00	1.00
Cesium Iodide (CsI)	<0.01	0.17	0.251
Tellurium Oxide (TeO2)	---	---	7.60E-03
Strontium Oxide (SrO)	---	---	4.10E-03
Molybdenum (MoO2)	---	---	6.60E-05
Cesium Hydroxide (CsOH)	---	---	1.00E-02
Barium Oxide (BaO)	---	---	1.90E-03
Lanthanides (La2O3)	---	---	3.80E-04
Cerium Oxide (CeO2)	---	---	2.70E-03
Antimony (Sb)	---	---	6.80E-02
Tellurium (Te2)	---	---	5.60E-04
Uranium Oxide (UO2)	---	---	1.20E-05

NOTES:

(1) All times referenced to t=0 (i.e., time of plant trip).

(2) End of Release based on MAAP graphical output and is taken as either: 1) for cases in which the releases taper off well before the end of the MAAP run, the point at which approximately 95% of the cumulative CsI release occurs (estimate based on visual inspection of output graph) or 2) otherwise, the end of the MAAP run (which is typically terminated 36 hours after vessel failure).

(3) Three points in time selected based on judgment and CsI release profile of the specific Fermi 2 MAAP case. Linear behavior is assumed between points in time.

(4) "---" indicates release fraction information not available for these radionuclides. Linearly increasing release fraction from 0.0 to value at End of Release assumed.

Table A-7

RADIONUCLIDE RELEASE FRACTIONS FOR EPRI CATEGORY #7d[1]

EPRI Category:	#7d - Containment Failure Due to Accident (d)		
Fermi MAAP Case:	III-C2		
Time of Initial Release:	1.8 hrs		
End of Release:[2]	36 hrs		
	Release Fraction to Environment as a Function of Time[3][4]		
Radionuclide	**@2 hrs**	**@20 hrs**	**@36 hrs**
Noble Gases	0.60	1.00	1.00
Cesium Iodide (CsI)	1.00E-02	2.50E-02	0.209
Tellurium Oxide (TeO2)	---	---	5.80E-02
Strontium Oxide (SrO)	---	---	2.70E-02
Molybdenum (MoO2)	---	---	5.10E-02
Cesium Hydroxide (CsOH)	---	---	7.90E-02
Barium Oxide (BaO)	---	---	4.60E-02
Lanthanides (La2O3)	---	---	3.60E-03
Cerium Oxide (CeO2)	---	---	1.60E-02
Antimony (Sb)	---	---	4.10E-01
Tellurium (Te2)	---	---	1.70E-02
Uranium Oxide (UO2)	---	---	8.70E-05

NOTES:

(1) All times referenced to t=0 (i.e., time of plant trip).

(2) End of Release based on MAAP graphical output and is taken as either: 1) for cases in which the releases taper off well before the end of the MAAP run, the point at which approximately 95% of the cumulative CsI release occurs (estimate based on visual inspection of output graph) or 2) otherwise, the end of the MAAP run (which is typically terminated 36 hours after vessel failure).

(3) Three points in time selected based on judgment and CsI release profile of the specific Fermi 2 MAAP case. Linear behavior is assumed between points in time.

(4) "---" indicates release fraction information not available for these radionuclides. Linearly increasing release fraction from 0.0 to value at End of Release assumed.

Table A-8

RADIONUCLIDE RELEASE FRACTIONS FOR EPRI CATEGORY #7e[1]

EPRI Category:	#7e - Containment Failure Due to Accident (e)		
Fermi MAAP Case:	IVA-1		
Time of Initial Release:	1.9 hrs		
End of Release:[2]	36 hrs		
	Release Fraction to Environment as a Function of Time[3][4]		
Radionuclide	**@6 hrs**	**@20 hrs**	**@36 hrs**
Noble Gases	1.00	1.00	1.00
Cesium Iodide (CsI)	4.00E-03	2.00E-02	6.30E-02
Tellurium Oxide (TeO2)	---	---	9.00E-03
Strontium Oxide (SrO)	---	---	2.90E-02
Molybdenum (MoO2)	---	---	2.20E-04
Cesium Hydroxide (CsOH)	---	---	1.40E-02
Barium Oxide (BaO)	---	---	1.30E-02
Lanthanides (La2O3)	---	---	3.40E-03
Cerium Oxide (CeO2)	---	---	1.30E-02
Antimony (Sb)	---	---	2.20E-01
Tellurium (Te2)	---	---	3.80E-03
Uranium Oxide (UO2)	---	---	8.80E-05

NOTES:

(1) All times referenced to t=0 (i.e., time of plant trip).

(2) End of Release based on MAAP graphical output and is taken as either: 1) for cases in which the releases taper off well before the end of the MAAP run, the point at which approximately 95% of the cumulative CsI release occurs (estimate based on visual inspection of output graph) or 2) otherwise, the end of the MAAP run (which is typically terminated 36 hours after vessel failure).

(3) Three points in time selected based on judgment and CsI release profile of the specific Fermi 2 MAAP case. Linear behavior is assumed between points in time.

(4) "---" indicates release fraction information not available for these radionuclides. Linearly increasing release fraction from 0.0 to value at End of Release assumed.

Table A-9

RADIONUCLIDE RELEASE FRACTIONS FOR EPRI CATEGORY #8[1]

EPRI Category:	#8 - Containment Bypass Accidents		
Fermi MAAP Case:	VX-1		
Time of Initial Release:	1 hrs		
End of Release:[2]	7 hrs		
	Release Fraction to Environment as a Function of Time[3][4]		
Radionuclide	**@2 hrs**	**@4 hrs**	**@7 hrs**
Noble Gases	0.90	1.00	1.00
Cesium Iodide (CsI)	0.210	0.215	0.217
Tellurium Oxide (TeO2)	---	---	7.80E-01
Strontium Oxide (SrO)	---	---	4.20E-02
Molybdenum (MoO2)	---	---	2.10E-02
Cesium Hydroxide (CsOH)	---	---	7.40E-01
Barium Oxide (BaO)	---	---	4.70E-02
Lanthanides (La2O3)	---	---	5.80E-03
Cerium Oxide (CeO2)	---	---	1.90E-02
Antimony (Sb)	---	---	7.50E-01
Tellurium (Te2)	---	---	2.80E-03
Uranium Oxide (UO2)	---	---	1.40E-04

NOTES:

(1) All times referenced to t=0 (i.e., time of plant trip).

(2) End of Release based on MAAP graphical output and is taken as either: 1) for cases in which the releases taper off well before the end of the MAAP run, the point at which approximately 95% of the cumulative CsI release occurs (estimate based on visual inspection of output graph) or 2) otherwise, the end of the MAAP run (which is typically terminated 36 hours after vessel failure).

(3) Three points in time selected based on judgment and CsI release profile of the specific Fermi 2 MAAP case. Linear behavior is assumed between points in time.

(4) "---" indicates release fraction information not available for these radionuclides. Linearly increasing release fraction from 0.0 to value at End of Release assumed.

A.3 GENERAL EMERGENCY DECLARATION

Another input required for the MACCS consequence calculations is an estimate of the time at which a General Emergency will be declared, per plant procedures, for each of the EPRI categories. The declaration of a General Emergency is used in the MACCS calculations as the initiator for offsite public evacuation.

The Fermi 2 MAAP calculations used to define the source terms are also used to define the time of General Emergency declaration.

The procedural guidance for declaration of a General Emergency is maintained at Fermi 2 in RERP Plan Implementing Procedure EP-101, Classification of Emergencies. A number of different criteria are defined in EP-101 for declaring emergency, the following three criteria are used to define the times of General Emergency declarations for the accidents scenarios of this analysis:

- SG2: Failure of the Reactor Protection System to Complete an Automatic Scram and Manual Scram was NOT Successful and there is Indication of an Extreme Challenge to the Ability to Cool the Core.
- FG1: Loss of Any Two Barriers AND Potential Loss of Third Barrier
- HG2: Other Conditions Existing Which in the Judgment of the Emergency Director Warrant Declaration of General Emergency

Criterion SG2 straight-forwardly applies to the ATWS scenario (MAAP Case IVA-1) used to characterize EPRI Category 7e. The conditions of this criterion are met in a few minutes for an unmitigated ATWS scenario.

Criterion FG1 applies to the short-term loss of coolant makeup accidents in this evaluation. Criterion FG1 considers the integrity of three fission product barriers: Fuel Clad Barrier; Reactor Coolant Barrier; and Primary Containment Barrier. A number of parameter thresholds are used to define Loss and Potential Loss of these barriers. The

key parameter in the case of the short term loss of coolant accidents of this risk assessment is RPV Water Level. Criterion FG1 defines Loss and Potential Loss of barriers in terms of water level, as follows:

Barrier	Loss	Potential Loss
Fuel Clad	< -40 inches	< 0 inches
Reactor Coolant	< 0 inches	n/a
Primary Containment	n/a	< -40 inches

Given these RPV water level thresholds of criterion FG1, a General Emergency declaration would be declared in approximately 15 minutes for the short term accident scenarios of this evaluation.

EP-101 criterion HG2 is used in the case of the long-term accident scenarios of this evaluation. Criterion HG2 allows Emergency Director discretion based on judgment that existing plant conditions may indicate actual or imminent substantial core degradation or melting with potential for loss of containment. If the discretion of the HG2 criterion did not exist in the EP-101 and criterion FG1 were relied upon to determine the point at which to declare a General Emergency, then the declaration time would be significantly delayed. For example, in the long-term loss of containment heat removal accident scenario modeled by Fermi 2 MAAP case IIA-2Z, the timings of key accident milestones are as follows (refer to Table A-1):

- Containment Failure 31.4 hrs
- Core Uncovery 34.3 hrs
- Core Damage 35.9 hrs

Using only EP-101 criterion FG1, a General Emergency would be delayed until 31.4 hrs (or later) into the event. With the discretion allowed the Emergency Directory by

criterion HG2, the Fermi Level 2 PRA (as well as this assessment) includes the interpretation that a General Emergency will be declared for loss of containment heat removal scenarios at about the time initiation of containment venting would be directed (approximately 15 hours into the scenario). At this point in such an accident scenario, the following conditions exist:

- DW pressure >1.68 psig (indicating loss of RCS Barrier)
- Containment design pressure approaching or exceeded, and opening of the emergency containment vent would have been attempted (indicating loss of Primary Containment Barrier)

Such conditions are judged here and in the Fermi PRA to warrant a reasonable assumption that the Emergency Director will invoke discretion (making a reasonable judgment that containment failure may severely impact continued core cooling) and declare a General Emergency.

The times of General Emergency declaration for each of the Fermi MAAP cases used to model the EPRI accident categories are summarized in Table A-1.

Appendix B
MACCS2 CONSEQUENCE CALCULATIONS

The severe accident consequence analysis was performed by Tetra Tech NUS, Inc. using the MACCS2 computer code. The Tetra Tech NUS, Inc. documentation of the Fermi ILRT MACCS2 calculations is provided in the following pages of this appendix.

Note that the EPRI category frequencies referenced in the attached calculation were preliminary values and do not match the final frequencies used in the Fermi 2 ILRT interval extension risk assessment; however, the population dose estimates are the final values.

TETRA TECH NUS, INC.

AIK-xx-xxxx
April 5, 2002

Vincent M. Andersen
ERIN Engineering & Research, Inc.
2105 S. Bascom Ave. #350
Campbell, CA 95008

SUBJECT: MACCS2 Calculations to Support Fermi ILRT Risk Assessment

Dear Mr. Anderson:

Tetra Tech NUS, Inc. is pleased to submit the attached letter report detailing the calculation of population doses to support the Fermi 2 ILRT frequency extension risk assessment. The MACCS2 code was used to estimate population doses based on the eight source terms and frequencies provided by ERIN Engineering and Research, Inc., and site-specific plant, demographic, agricultural, and meteorological data.

The resulting frequency-weighted collective dose to the population within 50 miles of the Fermi 2 plant was estimated to be 0.178 person-sieverts/year (17.8 person-rem/year).

If you have any questions, please contact me or Steve Connor at 803-649-7963.

Sincerely,

Ernesto Faillace
Project Manager

Attachment

cc: Steve Connor, TtNUS
Jon Cudworth, TtNUS
File: N4266-1.7

cc w/o att: D.M. Evans, TtNUS

MACCS2 Calculations To Support Fermi ILRT Risk Assessment

Introduction

This letter report was prepared by Tetra Tech NUS, Inc. To support the integrated leak rate testing (ILRT) frequency extension risk assessment for the Enrico Fermi Unit 2 (Fermi 2) nuclear power plant, located in Monroe County, Michigan. This risk assessment considers the annual frequency and potential source release fractions from accidents resulting in radionuclide releases from the Fermi 2 containment. Computer codes and methodologies approved by the nuclear regulatory commission (NRC), as well as site-specific demographic, agricultural, and meteorological data were used to assess the accident consequences from eight potential radionuclide source terms. The radiological consequence from each source term was multiplied by the annual frequency of each accident to derive a frequency-weighted dose to the population within 50 miles of Fermi 2.

Technical Approach

The severe accident consequence analysis was carried out with the MELCOR Accident Consequence Code System (MACCS2). MACCS2 simulates the impact of severe accidents at nuclear power plants on the surrounding environment. The principal phenomena considered in MACCS are atmospheric transport, mitigative actions based on dose projections, dose accumulation by a number of pathways including food and water ingestion, early and latent health effects, and economic costs. For purposes of the Fermi 2 ILRT Risk Assessment, economic costs were not considered. This analysis was performed with the MACCS2 version designated as Oak Ridge National Laboratory RSICC Computer Code Collection MACCS2 V.1.12, CCC-652 Code Package [1,2,3].

The input data sources, calculations, results, and sensitivity analysis are summarized in the next sections. The MACCS2 code is described in the technical reports provided with the code package.

Computer Codes And Computer Used

The codes used are MACCS2 [1] and SECPOP90 [4]. The calculations were carried out on a Dell PC computer. Spreadsheet calculations were done with Excel 97. Population calculations were performed using the ArcView geographical information system (GIS) software.

Input Data And Assumptions

Population Estimate

The year 2000 population within 50 miles of Fermi 2 was used as the basis for the collective dose estimate. The population within this radius was allocated to 160 sectors as a function of direction and distance from the plant. The sixteen azimuthal sections correspond to the sixteen compass directions used in the meteorological data input file. Ten annular sections were used to estimate the population at increasing distance from the site. These sections cover five one-mile increments from zero to five miles from the site, one five-mile increment from five to 10 miles, and four 10-mile intervals from 10 to 50 miles.

The US population within this 50-mile radius was estimated by downloading year 2000 block-level US Census data for Michigan and Ohio into a GIS database. These populations were apportioned based on the area that each census block

occupied within each sector that included US territory, assuming that the population in each block was uniformly distributed. The total year 2000 US population within 50 miles of Fermi 2 is estimated to be 5,180,871. The GIS also was used to estimate the proportion of each sector that was US territory, Canadian territory, or large bodies of water. The GIS was not used to estimate the fraction of each sector occupied by small bodies of water (reservoirs, rivers, etc.)

Due to the presence of a substantial Canadian population within the 50-mile radius, and the lack of current data concerning the distribution of the Canadian population, some adjustments were made to estimate the current Canadian population in any sector that included Canadian territory. This was done in several steps as follows:

1. Five sectors that included both US and Canadian territory were identified. These sectors are expected to contain both US and Canadian populations. A sixth sector was also identified as including very small fractions of both US and Canadian territory, but no population was registered in that sector. Eleven sectors included Canadian territory, but not US territory. The US Census-based GIS database produced no US populations in these sectors, as expected.
2. The year 2000 US populations estimated by the GIS database in each of the five dual-territory sectors was subtracted from the year 2000 population projected by the Fermi 2 UFSAR (Figure 2.1-9), which includes the Canadian population in those sectors.
3. The resulting populations in those five sectors were attributed to Canada and were summed to the populations projected to the year 2000 in Figure 2.1-9 of the UFSAR for the 11 Canadian sectors that did not included US territory. This resulted in a total (unadjusted) Canadian population of 394,627.
4. The 1970 populations in the Canadian towns and cities within 50 miles from Fermi 2 (from the UFSAR, Table 2.1-1) were summed, resulting in a total 1970 Canadian population of 234,918. The city of Windsor, Ontario, accounted for 200,790 people in 1970, and the remaining 34,128 were distributed between 10 towns
5. The current population of Windsor is estimated to be 307,877, based on the 2001 Canadian Census. This indicates that the Windsor population increased by 53 percent between the 1970 UFSAR estimate and 2001 Canadian Census estimate.
6. The same 53 percent growth rate was applied to the 10 Canadian towns, resulting in an estimated population of 52,329 in 2001. It is likely that this is an overestimate of the population in these 10 towns, as rural areas tend to grow at a slower rate than large metropolitan areas.
7. Summing the populations in steps 5 and 6 results in a 2001 Canadian population estimate of 360,206 within 50 miles of Fermi 2. This estimate is lower than the Canadian population calculated in step 3 by a factor of 0.913.
8. The scaling factor obtained in step 7 was applied to the population in each sector considered in step 3 to obtain a better estimate of the current Canadian population in those 16 sectors.
9. The scaled year 2001Canadian populations were combined with the year 2000 US populations in each of the sectors common to both countries. No correction was made for the difference of one year between the Canadian census and US census estimates. This will tend to slightly overestimate the year 2000 Canadian population, assuming that net population growth occurred between 2000 and 2001.
10. The total year 2000 population, including both US and Canadian populations within 50 miles of Fermi 2 was estimated to be 5,541,077.

Meteorological Parameters

The MACCS2 code requires a full year of hourly measurements of wind speed and direction, atmospheric stability and precipitation. The data used for this risk assessment were collected from the Fermi 2 on-site meteorological station at the 10-meter height during 1995. It was determined that this year represented the most complete data for the period 1995-1999. Only one hour of data was missing for all parameters except precipitation. Approximately 150 hours of precipitation data were missing in 1995. All other years had significantly greater gaps in the sequential meteorological data, with 1996 representing the next best year.

The one-hour gap in the wind speed, direction and stability data was patched by interpolating the parameters using the values for the preceding and following hour. The precipitation data gaps were patched by assuming the same rainfall recorded at Detroit Metropolitan Airport during the hours in which precipitation data were missing [5].
Tables Y-1 and Y-2 shows the resulting population distribution within 50 miles of Fermi 2

Table Y-1. Estimated population distribution within a 10-mile radius of Fermi 2, year 2000.

Sector	0-1 mile	1-2 miles	2-3 miles	3-4 miles	4-5 miles	5-10 miles	10-mile total
N	17	122	226	315	299	17,261	18,240
NNE	6	113	200	226	165	7,720	8,430
NE	1	40	99	3	1	434	578
ENE	0	0	0	0	0	0	0
E	0	0	0	0	0	0	0
ESE	0	0	0	0	0	0	0
SE	0	0	0	0	0	0	0
SSE	0	0	0	0	0	0	0
S	2	646	56	0	0	0	704
SSW	2	829	9	0	0	0	840
SW	2	68	2	26	247	2,365	2,710
WSW	3	95	837	2,659	1,498	32,748	37,840
W	14	98	163	293	506	6,444	7,518
WNW	21	98	173	239	1,229	4,566	6,326
NW	23	138	274	406	728	4,518	6,087
NNW	23	136	264	387	416	3,196	4,422
Total	113	2382	2302	4,553	5,088	79,247	93,685

Table Y-2. Estimated population distribution within a 50-mile radius of Fermi 2, year 2000.

Sector	0-10 miles	10-20 miles	20-30 miles	30-40 miles	40-50 miles	50-mile total
N	18,240	142,559	527,104	539,360	368,834	1,596,097
NNE	8,430	124,814	445,579	771,083	423,288	1,773,194
NE	578	14,207	82,677	85,973	1,432	184,867
ENE	0	8,305	18,773	24,200	19,042	70,320
E	0	348	3,021	11,917	685	15,971
ESE	0	0	0	482	0	482
SE	0	0	545	7,460	42,993	50,998
SSE	0	0	2,219	15,567	26,040	43,826
S	704	463	7,949	16,788	33,416	59,320
SSW	840	2,011	78,180	22,638	41,387	145,056
SW	2,710	20,698	316,223	104,496	24,767	468,894
WSW	37,840	10,153	15,527	9,956	12,526	86,002
W	7,518	8,620	8,313	48,105	20,667	93,223
WNW	6,326	5,930	24,504	13,945	17,948	68,653
NW	6,087	20,554	132,967	133,191	53,975	346,774
NNW	4,422	31,549	223,110	150,869	127,450	537,400
Total	93,685	390,201	1,886,681	1,956,020	1,214,440	5,541,077

The patched hourly data set was then converted to the format required for input to the MACCS2 code using an Excel spreadsheet. The final data items required for the meteorological input file, the morning and afternoon mixing heights, were estimated from US isopleth maps of mean annual mixing heights [6: Figures 1 and 6]. For Fermi 2, these values were estimated to be 510 m and 1200 m respectively.

Building Dimensions

The MACCS2 code requires the height and dimensions of the building from which the radionuclides would be released. The rectangular facade building that surrounds the containment building and the auxiliary building has the width dimensions of 160 ft in the N-S direction and 209 ft in the E-W direction [7]. The reactor building itself is 152 ft above grade. For the purposes of these analyses, it is assumed that the leak/rupture of the containment does not destroy the facade building so that the sigma values for calculating diffusion will be those of the facade. The average of the N-S and E-W widths (185 ft) is selected for use here, so that the MACCS2 input for containment width is 56.3 m and the height is 46.4 m.

Plume Release Height

The postulated releases from the Primary Containment proceed in almost all cases up through the Reactor Building, through the hatch in the Refuel Floor, and out the various blowout panels in the Refuel Floor walls (approximately 152 ft above grade). Therefore, the release height in this risk assessment is assumed to be at 46.4 m. Buoyant plume rise was conservatively assumed to be zero.

Reactor Core Inventory

The MACCS2 code data includes a 3412 MWTH PWR radionuclide isotopic core inventory at end-of-cycle. This is scaled to approximate the Fermi 2 inventory by the ratio of thermal powers. The thermal power of the Fermi 2 reactor is 3430 MWth so that the scaling factor is 1.005 (=3430/3412).

Source Term

Eight release scenarios, derived from the Fermi 2 MAAP runs, are summarized in Table X-1 of Ref. [8]. In order to convert from 12 MAAP release fractions to 9 MACCS2 release fractions, the following assumptions were made, showing the MACCS2 category first:

- Xe/Kr - Noble Gases
- I - CsI
- Cs - CsOH
- Te - Maximum of TeO2, Te2, or Sb
- Sr - SrO
- Ru - MoO2
- La - La2O3
- Ce - CeO2
- Ba - BaO

Table Y-3 summarizes the times at which a general emergency is declared and when the release is modeled to start and end. Three plumes are sequentially tracked for each release category, taking into account the release fractions as a function of time based on the information contained in Tables X-4 through X-11 of Ref. [8]. Some minor modifications in plume duration were made for EFILRT-2 and EFILRT-3, since MACCS2 cannot handle plume durations exceeding 24 hours. These modifications have the effect of slightly overestimating the doses. Due to problems that MACCS2 has with four or more plumes per source, it was not practical to split the longer plume segments by adding a fourth plume. The duration and release fractions for each plume segment are listed in Tables Y-4 through Y-11.

Table Y-3. Time of general emergency declaration and timing of releases (in seconds from time of plant trip).

MACCS2 Run	EFILRT-1	EFILRT-2	EFILRT-3	EFILRT-4	EFILRT-5	EFILRT-6	EFILRT-7	EFILRT-8
General Emergency	1080	1080	54000	54000	54000	1080	360	1440
Start of Release	5400	6480	133200	101520	133200	6480	6840	3600
End of Release	129600	104400*	316800*	187200	234000	129600	129600	25200

*Shortened due to MACCS2 limitations on plume duration.

Table Y-4. Radionuclide release fractions for source term EFILRT-1.

	Duration (seconds)	XE/KR	I	CS	TE	SR	RU	LA	CE	BA
Plume 1	1800	7.25E-05	5.80E-09	5.80E-09	2.90E-10	0.00E+00	7.25E-12	0.00E+00	0.00E+00	4.35E-12
Plume 2	64800	2.61E-03	2.09E-07	2.09E-07	1.04E-08	0.00E+00	2.61E-10	0.00E+00	0.00E+00	1.57E-10
Plume 3	57600	2.32E-03	1.86E-07	1.86E-07	9.28E-09	0.00E+00	2.32E-10	0.00E+00	0.00E+00	1.39E-10
Total	124200	5.00E-03	4.00E-07	4.00E-07	2.00E-08	0.00E+00	5.00E-10	0.00E+00	0.00E+00	3.00E-10

Table Y-5. Radionuclide release fractions for source term EFILRT-2.

	Duration (seconds)	XE/KR	I	CS	TE	SR	RU	LA	CE	BA
Plume 1	720	5.00E-01	1.00E-02	7.35E-05	5.00E-04	3.01E-05	4.85E-07	2.79E-06	1.99E-05	1.40E-05
Plume 2	10800	5.00E-01	1.40E-01	1.10E-03	7.50E-03	4.52E-04	7.28E-06	4.19E-05	2.98E-04	2.10E-04
Plume 3	86400*	0.00E+00	1.01E-01	8.82E-03	6.00E-02	3.62E-03	5.82E-05	3.35E-04	2.38E-03	1.68E-03
Total	97920	1.00E+00	2.51E-01	1.00E-02	6.80E-02	4.10E-03	6.60E-05	3.80E-04	2.70E-03	1.90E-03

*Release assumed to stop at 29 hours, rather than 36 hours from reactor trip.

Table Y-6. Radionuclide release fractions for source term EFILRT-3.

	Duration (seconds)	XE/KR	I	CS	TE	SR	RU	LA	CE	BA
Plume 1	10800	8.00E-01	5.00E-05	2.41E-06	1.12E-06	0.00E+00	1.18E-10	0.00E+00	0.00E+00	2.76E-11
Plume 2	86400*	2.00E-01	2.00E-05	1.93E-05	8.94E-06	0.00E+00	9.41E-10	0.00E+00	0.00E+00	2.21E-10
Plume 3	86400*	0.00E+00	9.00E-05	1.93E-05	8.94E-06	0.00E+00	9.41E-10	0.00E+00	0.00E+00	2.21E-10
Total	183600	1.00E+00	1.60E-04	4.10E-05	1.90E-05	0.00E+00	2.00E-09	0.00E+00	0.00E+00	4.70E-10

*Plume 2 was shortened from 30 to 24 hours and Plume 3 was extended from 20 to 24 hours. Release is assumed to stop at 88 hours, rather than 90 hours from reactor trip.

Table Y-7. Radionuclide release fractions for source term EFILRT-4.

	Duration (seconds)	XE/KR	I	CS	TE	SR	RU	LA	CE	BA
Plume 1	6480	1.00E+00	1.50E-03	1.29E-04	2.72E-03	1.89E-04	1.44E-09	6.96E-06	5.37E-05	8.32E-05
Plume 2	36000	0.00E+00	2.40E-03	7.14E-04	1.51E-02	1.05E-03	7.98E-09	3.87E-05	2.98E-04	4.62E-04
Plume 3	43200	0.00E+00	1.00E-04	8.57E-04	1.82E-02	1.26E-03	9.58E-09	4.64E-05	3.58E-04	5.55E-04
Total	85680	1.00E+00	4.00E-03	1.70E-03	3.60E-02	2.50E-03	1.90E-08	9.20E-05	7.10E-04	1.10E-03

Table Y-8. Radionuclide release fractions for source term EFILRT-5.

	Duration (seconds)	XE/KR	I	CS	TE	SR	RU	LA	CE	BA
Plume 1	10800	5.00E-01	1.00E-02	1.07E-03	7.29E-03	4.39E-04	7.07E-06	4.07E-05	2.89E-04	2.04E-04
Plume 2	18000	5.00E-01	1.60E-01	1.79E-03	1.21E-02	7.32E-04	1.18E-05	6.79E-05	4.82E-04	3.39E-04
Plume 3	72000	0.00E+00	8.10E-02	7.14E-03	4.86E-02	2.93E-03	4.71E-05	2.71E-04	1.93E-03	1.36E-03
Total	100800	1.00E+00	2.51E-01	1.00E-02	6.80E-02	4.10E-03	6.60E-05	3.80E-04	2.70E-03	1.90E-03

Table Y-9. Radionuclide release fractions for source term EFILRT-6.

	Duration (seconds)	XE/KR	I	CS	TE	SR	RU	LA	CE	BA
Plume 1	720	6.00E-01	1.00E-02	4.62E-04	2.40E-03	1.58E-04	2.98E-04	2.11E-05	9.36E-05	2.69E-04
Plume 2	64800	4.00E-01	1.50E-02	4.16E-02	2.16E-01	1.42E-02	2.68E-02	1.89E-03	8.42E-03	2.42E-02
Plume 3	57600	0.00E+00	1.84E-01	3.70E-02	1.92E-01	1.26E-02	2.39E-02	1.68E-03	7.49E-03	2.15E-02
Total	123120	1.00E+00	2.09E-01	7.90E-02	4.10E-01	2.70E-02	5.10E-02	3.60E-03	1.60E-02	4.60E-02

Table Y-10. Radionuclide release fractions for source term EFILRT-7.

	Duration (seconds)	XE/KR	I	CS	TE	SR	RU	LA	CE	BA
Plume 1	14760	1.00E+00	4.00E-03	1.68E-03	2.65E-02	3.49E-03	2.65E-05	4.09E-04	1.56E-03	1.56E-03
Plume 2	50400	0.00E+00	1.60E-02	5.75E-03	9.03E-02	1.19E-02	9.03E-05	1.40E-03	5.34E-03	5.34E-03
Plume 3	57600	0.00E+00	4.30E-02	6.57E-03	1.03E-01	1.36E-02	1.03E-04	1.60E-03	6.10E-03	6.10E-03
Total	122760	1.00E+00	6.30E-02	1.40E-02	2.20E-01	2.90E-02	2.20E-04	3.40E-03	1.30E-02	1.30E-02

Table Y-11. Radionuclide release fractions for source term EFILRT-8.

	Duration (seconds)	XE/KR	I	CS	TE	SR	RU	LA	CE	BA
Plume 1	3600	9.00E-01	2.10E-01	1.23E-01	1.30E-01	7.00E-03	3.50E-03	9.67E-04	3.17E-03	7.83E-03
Plume 2	7200	1.00E-01	5.00E-03	2.47E-01	2.60E-01	1.40E-02	7.00E-03	1.93E-03	6.33E-03	1.57E-02
Plume 3	10800	0.00E+00	2.00E-03	3.70E-01	3.90E-01	2.10E-02	1.05E-02	2.90E-03	9.50E-03	2.35E-02
Total	21600	1.00E+00	2.17E-01	7.40E-01	7.80E-01	4.20E-02	2.10E-02	5.80E-03	1.90E-02	4.70E-02

Evacuation Fraction and Speed

The MACCS2 Users Guide input parameters of 95 percent of the population within 10 miles of the plant (Emergency Planning Zone) evacuating and 5 percent not evacuating were employed. These values have been used in similar studies (e.g., [9], [10]) and are conservative relative to the NUREG-1150 study, which assumed evacuation of 99.5 percent of the population within the emergency planning zone [11].

The Fermi 2 evacuation time estimate analysis [12] presents various emergency plan evacuation scenarios for each of 35 "Protective Action Areas." The evacuation time estimates for each area were developed for eight combinations of conditions, including: summer and winter seasons, daytime and nighttime, and normal and adverse weather variables. The results indicate that there is little difference between summer and winter evacuation estimates. Under normal weather conditions, the general evacuation time estimates for the full Emergency Planning Zone (EPZ, 0-10 miles) range from 3.25 hours during the day and 2.42 hours at night. Under adverse weather conditions, the evacuation time estimates range from 3.92 hours during the day to 2.67 hours at night.

The above data have been modeled here by assuming an average radial evacuation speed of 2.6 miles per hour (10 miles/3.92 hours) , i.e. 1.14 meters per second, as representing the slowest evacuation range (in adverse daytime winter weather) in various directions. Evacuation was assumed to begin 15 minutes after the general alarm is sounded.

Land Fractions

The SECPOP90 code [4] calculates the land fraction for each rosette section as explained in the manual for the code. The code contains a county-level database with the land fractions for each county obtained from the 1990 census data files. The calculated values for the sectors that did not touch any lake area are used directly in this analysis. Due to the way in which SECPOP90 allocates population from the census blocks, certain of the radial blocks near the plant are shown as all water. Therefore, the land fractions for any sectors that included lake water were calculated using the GIS software.

Watershed Index and Definition

The sectors were assigned to one river watershed for the land sectors and to a lake watershed for the predominately Great Lakes area sectors. The River watershed is assumed to have the uptake values listed in the NUREG-1150 study [11]. The Lake watershed is assumed to have zero uptake due to the very large amount of dilution.

Crop Season and Share

Agricultural production information was taken from the 1997 USDA Agricultural Census [13]. Production within 50 miles of the site was estimated based on the 18 US counties within this radius. Production in those counties that lie partially outside of this area was scaled by the fraction of the county within the area of interest. No significant amounts of cropland in Michigan and Ohio are dedicated to crops that are not for human or animal food consumption (e.g., tobacco and cotton). Of the food crops, grains (46 percent of the total cropland, mostly corn and wheat) and legumes (43 percent of total cropland, primarily soybeans) were harvested from the largest areas. Other crops consumed less than 8 percent of the croplands within 50 miles of the site; pasture made up close to 4 percent of this land.

The duration of the growing seasons for the principal grains and legumes were obtained from the USDA for crops grown in Michigan and Ohio [14]. The duration of the growing season for the remaining crop categories (pasture, stored forage, green leafy vegetables, roots/tubers, and other food crops) were based on reasonable estimates of growing periods for cold-weather latitudes. The uncertainty associated with these estimates is of minor impact since they occupy a relatively small fraction of the total cropland.

Economic Parameters

The economic parameters generated by the SECPOP90 code were not updated since no economic impacts are considered in this risk assessment. They were included in the site input file to ensure that MACCS2 would properly execute, but the resulting output was ignored.

Results

The MACCS2 population dose results are given in tables of the effective dose equivalent for various statistical measures, including the mean, 50th, 95th, 99th, and 99.5th percentiles, as well as the peak consequence. These doses are reported for the population within 10 and 50 miles of the site. This analysis reports the mean collective doses (in person-sieverts) to the population within 50 miles of Fermi 2 calculated by the MACCS2 code for each of the eight source terms. The collective dose is then multiplied by the release frequency based on the EPRI category for each source term. The frequency-weighted doses are then summed to obtain the total weighted dose for all ILRT source terms. As indicated in Table Y-12, the mean frequency-weighted annual population dose for this ILRT risk assessment is estimated to be 0.178 person-sieverts/year (17.8 person-rem/year).

Table Y-12. Mean and frequency-weighted collective dose to the 50-mile population

ILRT Source Term	Dose (person-Sv)	Frequency (y^{-1})	Wt. Dose (person-Sv/y)
ILRT-1	2.05E+00	1.43E-08	2.93E-08
ILRT-2	4.09E+04	1.00E-11	4.09E-07
ILRT-3	1.88E+02	1.95E-07	3.67E-05
ILRT-4	1.25E+04	2.48E-06	3.10E-02
ILRT-5	4.31E+04	2.33E-07	1.00E-02
ILRT-6	1.21E+05	1.73E-07	2.09E-02
ILRT-7	7.41E+04	1.40E-06	1.04E-01
ILRT-8	1.75E+05	7.20E-08	1.26E-02
Total	N/A	4.57E-06	1.78E-01

Sensitivity Analysis

The base case made use of the meteorological data set collected during 1995 and conservatively assumed that five percent of the population would not evacuate. Two separate runs were made to determine the sensitivity of the calculated doses to changes in the meteorological data set used as well as the evacuation assumption.

Meteorological Data Set

An analysis was performed to determine whether the results were sensitive to the year in which the meteorological data were collected. The meteorological data collected during 1996 were used for this comparison, since this was the second most complete data set available. The gaps in the 1996 data set were patched by interpolation and/or substitution with 1997 data for the same time periods. Missing precipitation data were taken from the rainfall amounts reported at Detroit Metropolitan Airport during the same period [5].

The change in the meteorological data set had a small, but not significant impact on the calculated doses. The ILRT-7 source term resulted in the greatest frequency-weighted collective dose in the base case. In the sensitivity run, the difference in the collective dose from the ILRT-7 source term was about two percent (7.58E+04 person-sieverts vs. 7.41E+04 person-sieverts). The total frequency-weighted dose did not change by more than three percent (1.83E-01 person-sieverts/year vs. 1.78E-01 person-sieverts/year).

Evacuation Assumptions

The assumption that five percent of the population within 10 miles of the plant would not evacuate was revised by assuming 100 percent evacuation. This assumption led to very small reductions (if any) in the collective doses relative to the base case results for some source terms. The frequency-weighted collective dose was essentially unchanged under the complete evacuation scenario.

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