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10 CFR 50.90

5928-06-20531
September 15, 2006

U.S. Nuclear Regulatory Commission
ATTN: Document Control Desk
Washington, DC 20555

Three Mile Island, Unit 1
Facility Operating License No. DPR-50
NRC Docket No. 50-289

Subject: Technical Specifications Change Request No. 334 - One-Time Type A Test Interval Extension

Pursuant to 10 CFR 50.90, AmerGen Energy Company, LLC (AmerGen) hereby requests an amendment to Appendix A, Technical Specifications, of Facility Operating License No. DPR-50. The proposed change modifies Technical Specifications (TS) 6.8.5, "Reactor Building Leakage Rate Testing Program." Specifically, the proposed change will revise TS 6.8.5 to reflect a one-time extension to the Three Mile Island, Unit 1 Type A Integrated Leak Rate Test (ILRT) interval as currently specified in the Technical Specifications. This change will extend the requirement to perform the Type A ILRT from the current requirement of "no later than September 2008" to "prior to startup from the T1R18 refueling outage" which is currently scheduled for Fall 2009.

AmerGen requests approval of the proposed changes by June 30, 2007. Once approved, the amendment shall be implemented within 60 days. The proposed changes have been reviewed by the Plant Operations Review Committee and approved by the Nuclear Safety Review Board. No new regulatory commitments are established by this submittal.

We are notifying the Commonwealth of Pennsylvania of this application for changes to the Technical Specifications by transmitting a copy of this letter and its attachments to the designated State Official.

If any additional information is needed, please contact Tom Loomis at (610) 765-5510.

I declare under penalty of perjury that the foregoing is true and correct. Executed on the 15th day of September, 2006.

Respectfully,

Pamela B. Cowan

Pamela B. Cowan
Director, Licensing & Regulatory Affairs
AmerGen Energy Company, LLC

A017

Attachments: (1) Evaluation of Proposed Change
(2) Markup of Proposed Technical Specification Page Change
(3) Retyped Page for Technical Specification Change
(4) Risk Assessment for TMI Unit 1 to Support ILRT (Type A) Interval Extension Request

cc: R. R. Janati, Commonwealth of Pennsylvania
S. J. Collins, Administrator, Region 1, USNRC
D. M. Kern, USNRC Senior Resident Inspector
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File No. 06050

ATTACHMENT 1

Evaluation of Proposed Change

CONTENTS

SUBJECT: One-Time Type A Test Interval Extension

1.0 INTRODUCTION

This letter is a request to amend Facility Operating License No. DPR-50 for Three Mile Island (TMI), Unit 1. The proposed change will revise TS 6.8.5 to reflect a one-time extension to the Three Mile Island, Unit 1 Type A Integrated Leak Rate Test (ILRT) interval as currently specified in the Technical Specifications.

This change will extend the requirement to perform the Type A ILRT from the current requirement of "no later than September 2008" to "prior to startup from the T1R18 refueling outage," which is currently scheduled for Fall 2009. This deferral would add approximately 15 months to the currently approved 15-year interval. This deferral will allow performance of the Type A ILRT during a planned steam generator replacement outage (T1R18) in the Fall 2009. The last TMI, Unit 1 Type A test was performed in September 1993. Based on the current wording of TS 6.8.5, the next Type A ILRT test is required to be performed no later than September 2008. This would result in performing the Type A test at the next scheduled refueling outage, which is T1R17 currently scheduled for Fall 2007, otherwise, a mid-cycle outage in 2008 would be necessary to perform the Type A ILRT. Compliance with the current TS would require performing the Type A ILRT test following conclusion of the T1R17 outage (currently scheduled for fall 2007) <u>and</u> additional containment pressure testing would be required at the conclusion of T1R18 (currently scheduled for fall 2009) as a result of the steam generator replacement outage.

AmerGen Energy Company, LLC (AmerGen) requests approval of the proposed change by June 30, 2007. Once approved, the amendment shall be implemented within 60 days.

2.0 PROPOSED CHANGE

The proposed change will revise TS 6.8.5 to reflect a one-time extension to the Three Mile Island, Unit 1 Type A ILRT as currently specified in the Technical Specifications.

The proposed change would revise Section 6.8.5 ("Reactor Building Leakage Rate Testing Program") of the TMI, Unit 1 Technical Specifications to state the following:

> "... as modified by the following exception to NEI 94-01, Rev. 0, 'Industry Guideline for Implementing Performance-Based Option of 10 CFR Part 50, Appendix J':
>
> a. Section 9.2.3: The first Type A test performed after the September 1993 Type A test shall be performed <u>prior to startup from the T1R18 refueling outage</u>."

3.0 BACKGROUND

TMI, Unit 1 is a Babcock and Wilcox pressurized water reactor with a large volume, dry containment structure. The design of the containment structure is discussed in the Reference 2 License Amendment Request.

The integrity of the containment penetrations and isolation valves is verified through Type B and Type C Local Leak Rate Tests (LLRTs) and the overall leak tight integrity of the containment is verified by a Type A ILRT, as required by 10 CFR 50, Appendix J, "Primary Reactor Containment Leakage Testing for Water-Cooled Power Reactors." These tests are performed to verify the essentially leak tight characteristics of the containment at the design basis accident pressure. The most recent Type A ILRT for TMI, Unit 1 was performed in September 1993.

Option B, "Performance-Based Requirements," of Appendix J to 10 CFR 50 requires that a Type A ILRT be conducted at a periodic interval based on historical performance of the overall containment system. TMI, Unit 1 TS 6.8.5 requires that a program be established to comply with the primary containment leakage rate testing requirements of 10 CFR 50.54(o) and 10 CFR 50, Appendix J, Option B, as modified by approved exemptions. Additionally, this program is in accordance with the guidelines contained in Regulatory Guide (RG) 1.163, "Performance-Based Containment Leak-Test Program," dated September 1995. 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 50, Appendix J," dated July 26, 1995 (Reference 1).

NEI 94-01 specifies an initial test interval of 48 months for a Type A ILRT and allows an extension of the interval to 10 years based on two consecutive successful tests.

The NRC has previously approved a one-time 5-year extension to the Type A ILRT test interval. In Reference 2, AmerGen submitted a License Amendment Request (LAR) to defer the Type A ILRT schedule for TMI, Unit 1. The TS were changed to state that the specified test shall be performed no later than September, 2008. In response to an NRC Request for Additional Information, AmerGen provided additional information regarding the LAR in References 3 and 4. The NRC approved the LAR in Reference 5.

As stated previously, compliance with the current TS would require performing the Type A test following conclusion of the T1R17 outage (currently scheduled for fall 2007) and additional containment pressure testing would be required at the conclusion of T1R18 (currently scheduled for fall 2009) as a result of the planned steam generator replacement outage. The proposed change would defer the Type A ILRT by approximately 15 months and avoid the duplication and additional resources necessary to perform two (2) containment pressurization tests. Performing a Type A ILRT during T1R17 and performing additional containment pressure testing at the conclusion of the T1R18 (currently scheduled for fall 2009) as a result of the planned steam generator replacement outage results in an imposed hardship without a significant increase in safety. The Type A ILRT results in a minimum 36-hour increase in outage duration and results in direct costs of approximately $375,000. Containment pressurization to satisfy ASME Boiler & Pressure Vessel Code Subsection IWL-5000 requirements in 2009 would result in a minimum 21-hour increase in outage duration and results in additional direct costs of approximately $150,000 for this outage.

A review of the anticipated activities associated with replacement of the steam generators was performed in order to identify the need for a Type A ILRT. The steam generator replacement will result in an anticipated containment breach of approximately 21 feet wide by 25 feet high. This breach will create a concrete and steel opening. Following repair of the breach, a local leak rate test of the liner repair weld will need to be performed in lieu of a Type A test. However, containment pressurization to the design basis accident pressure of the containment would still be necessary to ensure the structural integrity of the replaced concrete and steel structure as required by ASME Boiler & Pressure Vessel Code Subsection IWL-5000. Therefore, containment pressurization to the design basis accident pressure would also be necessary during the planned steam generator replacement outage. This additional containment pressurization during the steam generator replacement outage would still involve significant manpower resources and costs such as rental of pressurization equipment, alignment of plant systems for containment closure, defeating pressure signals, protecting balance of plant equipment inside containment from the effects of the pressure, and preventing access to the containment for an extended period of time.

Accordingly, the proposed change would defer the Type A ILRT by approximately 15 months and avoid the duplication and additional resources necessary to perform two (2) containment pressure tests within 24 months. Performing containment pressure tests during two (2) consecutive outages would result in an imposed hardship without a corresponding increase in safety. As discussed below, this submittal demonstrates that there is a small change in risk to the public based on extending the required Type A ILRT test from September, 2008 to the T1R18 (Fall 2009) refueling outage.

4.0 TECHNICAL ANALYSIS

4.1 10CFR 50, Appendix J, Option B

The testing requirements of 10 CFR 50, Appendix J provide assurance that leakage through the containment, including systems and components that penetrate the containment, does not exceed allowable leakage rate values specified in the TS and Bases. The allowable leakage rate is limited such that the leakage assumptions in the safety analyses are not exceeded. The limitation of containment leakage provides assurance that the containment would perform its design function following an accident, up to and including the design basis accident.

10 CFR 50, Appendix J was revised, effective October 26, 1995, to allow licensees to choose containment leakage testing under Option A, "Prescriptive Requirements," or Option B. Amendment No. 201 for TMI, Unit 1 was issued to permit implementation of 10 CFR 50, Appendix J, Option B. TS 6.8.5 currently requires the establishment of a leakage testing program in accordance with 10 CFR 50.54(o) and 10 CFR 50, Appendix J, Option B, as modified by approved exemptions. This program implements the guidelines contained in RG 1.163 which specifies a method acceptable to the NRC for complying with Option B by approving the use of NEI 94-01, subject to several regulatory positions stated in RG 1.163.

10 CFR 50, Appendix J, Option B, Section V.B.3 specifies that RG 1.163, or other implementing documents used to develop a performance-based leakage testing program must be included, by general reference, in the plant's TS. Additionally, deviations from guidelines endorsed in a regulatory guide are to be submitted as a revision to the plant's TS. Therefore, this application does not require an exemption from 10 CFR 50, Appendix J, Option B.

The adoption of the Option B performance-based containment leakage rate testing program did not alter the basic method by which Appendix J leakage rate testing is performed or its acceptance criteria, but it did alter the test frequency of containment leakage testing in Type A, B, and C tests. The required testing frequency is based upon an evaluation which utilizes the "as-found" leakage history to determine the frequency for leakage testing which provides assurance that leakage limits will be maintained.

The allowable frequency for the Type A ILRT is based, in part, upon a generic evaluation documented in NUREG-1493, "Performance-Based Containment Leak-Test Program" (Reference 6). NUREG-1493 made the following observations with regard to changing the test frequency:

- Reducing the Type A ILRT frequency to once per twenty years was found to lead to an imperceptible increase in risk. The estimated increase in risk is small because Type A ILRTs identify only a few potential leakage paths that cannot be identified by Type B and C testing, and the leaks that have been found by Type A ILRTs have only been marginally above the existing requirements. Given the insensitivity of risk to containment leakage rate, and the small fraction of leakage detected solely by Type A ILRTs, increasing the interval between Type A ILRTs has minimal impact on public risk.

- While Type B and C tests identify the vast majority (i.e., greater than 95%) of all potential leakage paths, performance-based alternatives are feasible without significant risk impacts. Since leakage contributes less than 0.1 percent of overall risk under existing requirements, the overall effect is very small.

The required surveillance frequency for Type A ILRTs in NEI 94-01 is at least once per ten years based on an acceptable performance history (i.e., two consecutive periodic Type A ILRTs at least 24 months apart or refueling cycles where the calculated performance leakage rate was less than 1.0 L_a). In August 2003, the NRC approved a one-time deferral of the Type A ILRT schedule for TMI, Unit 1 (Reference 5). The schedule deferral provided a test date consistent with a 15-year test schedule. For TMI, Unit 1, the specified test date was changed to "no later than September 2008."

4.2 Integrated Leak Rate History

Type A ILRT testing is performed to verify the integrity of the containment structure. Industry test experience has demonstrated that Type B and C tests detect a large percentage of containment leakage and that the percentage of containment leakage detected only by integrated containment leakage testing is very small. Results of previous TMI, Unit 1 Type A ILRTs demonstrate that the TMI, Unit 1 containment structure remains essentially a leak tight barrier and represents minimal risk to increased leakage. These plant specific results support the conclusions of NUREG-1493. The specific results from the previous Type A ILRTs are contained in the Reference 2 License Amendment Request.

4.3 Containment Inspections

As discussed in the Reference 2 submittal, TMI, Unit 1 is committed to the 1992 Edition with the 1992 Addenda of the ASME Boiler and Pressure Vessel Code, Section XI.

In accordance with the 10 CFR 50.55a(b)(2)(ix)(E) a General Visual Examination of the Reactor Building containment liner must be performed during each Section XI ISI period of the ten-year interval. TMI, Unit 1 has completed this exam for the first period and is scheduled to complete the second period examination in 2007. TMI, Unit 1 is also required to perform a VT-3 (Item No. E1.12) of the Reactor Building containment liner of the accessible liner courses. This examination is required to be performed during the third period of this 10-year interval and is currently scheduled for completion in 2011. An additional visual exam of the area adjacent to the moisture barrier (i.e., between liner and concrete) is performed each refueling outage.

Additional information concerning the inspection of the containment liner and the moisture barrier interface was provided in the Reference 3 submittal. As discussed in the Reference 3 submittal, during T1R13 (1999), 100% of the accessible portions of the containment building liner and moisture barrier interface were examined in accordance with Section XI, Examination Category E-A, E1.11. Additionally, an augmented exam was conducted in accordance with the ASME Section XI IWE, Examination Category E-C, Item Number E4.12, which requires a volumetric wall thickness examination at selected areas to be performed each period. This examination identified five (5) areas where coating degradation had resulted in localized liner metal loss of 1/16" at the liner-to-concrete slab and moisture barrier (seal) interface (Elevation 281'). It was determined by Engineering through calculation, that the remaining liner wall thickness would afford the necessary containment membrane/lining to ensure leak tightness. In order to preserve the remaining liner wall thickness coating repairs were completed during T1R13 (1999), and a supplemental exam (VT-1) of these areas was completed during T1R14 (2001). In T1R14, a supplemental VT-1 examination was completed and the five (5) areas were found to be acceptable. As an additional action for the augmented examination requirements, six (6) areas were marked for UT examination from the 281' elevation inside the Reactor Building. The liner thickness of these six (6) areas was found to be consistent with little to no wall loss. These six areas are scheduled for UT examination again during the 2007 refueling outage.

Visual examination of 360 degrees of the circumference of the moisture barrier region in 2003 identified areas of chipped, scratched, and pealed paint and broken moisture barrier sealing. An area of corrosion was identified and UT examined to determine remaining liner thickness. No areas less than 0.300" were identified. These areas were corrected and re-examined to assure they were corrected. A similar 360 degree visual examination of the moisture barrier region was completed again during the 2005 refueling outage. This examination identified six areas of the moisture barrier region where the caulking to the liner was no longer bonded and required repair. These areas were corrected and subsequently examined.

As also discussed in the Reference 3 letter, there was one (1) area where a round indication 1/16" x 1/16" deep was observed at Elev. 374' in the liner during T1R13 (1999). This indication was not the result of corrosion. It was determined by Engineering through calculation that the remaining liner wall thickness would afford the necessary containment membrane/lining to ensure leak tightness. A supplemental exam (VT-1) of this area was specified for T1R14 (2001 outage). The supplemental exam (VT-1) was performed and the area found to be acceptable.

Containment inspections also include an examination of pressure retaining bolting. Pressure retaining bolting examinations have been completed on 16 bolted penetration connections. These examinations were performed in accordance with ASME Section XI, Examination Category E-G, Item No. E8.10. All of the bolted penetration connections are currently scheduled for visual examination in 2007 in order to coincide with the scheduled IWE General Visual examination required for the examination period.

In summary, the containment liner areas which had experienced some degradation were identified, analyzed and repaired as necessary to ensure an acceptable containment barrier still exists.

4.4 Risk Analysis

As discussed in Attachment 4, the Probabilistic Safety Risk Assessment results demonstrate a very small impact on risk associated with the one time extension of the containment Type A integrated leak rate test (ILRT) interval from 15 years to 16.25 years (an additional 15-month extension). The risk assessment follows the guidelines of EPRI TR-1009325, "Risk Impact Assessment of Extended Integrated Leak Rate Testing Intervals," Revision 1, dated December 2005 (Reference 7), and the methodology outlined in a response to a Request for Additional Information for a previous TMI, Unit 1 ILRT submittal (Reference 3), which used an approach similar to what was done in a study for Calvert Cliffs to estimate the likelihood and risk implications of corrosion-induced leakage of steel liners going undetected during the extended test interval. The risk assessment is also based on the interim NEI guidance concerning Type A ILRT extensions (Reference 9).

The following is a brief summary of some of the key aspects of the Type A ILRT test interval extension risk analysis for a one-time increase from 15 years to 16.25 years:

- Increasing the current one-time 15-year ILRT interval to 16.25 years results in an insignificant increase in total population dose rate, from 10.750 person-rem/year to 10.768 person-rem/year, respectively.

- The increase in the LERF risk measure, 1.45E-8/yr, is categorized as a "very small" increase per NRC Regulatory Guide 1.174 (Reference 8).

- Likewise, the conditional containment failure probability ($CCFP_{\%}$) increases insignificantly by 0.1 percentage points.

5.0 REGULATORY ANALYSIS

5.1 NO SIGNIFICANT HAZARDS CONSIDERATION

AmerGen Energy Company, LLC (AmerGen) has evaluated the proposed change to the Technical Specifications (TS) for Three Mile Island (TMI), Unit 1 and has determined that the proposed changes do not involve a significant hazards consideration and is providing the following information to support a finding of no significant hazards consideration.

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

Response: No

The proposed change will revise TS 6.8.5 to reflect a one-time extension to the Three Mile Island, Unit 1 Type A Integrated Leak Rate Test (ILRT) as currently specified in the Technical Specifications. This change will extend the requirement to perform the Type A ILRT from the current requirement of "no later than September 2008" to "prior to startup from the T1R18 refueling outage," which is currently scheduled for Fall 2009. The current Type A ILRT interval of 15 years, based on past performance, would be extended on a one-time basis by approximately 15 months.

The function of the containment is to isolate and contain fission products released from the reactor coolant system following a design basis Loss of Coolant Accident (LOCA) and to confine the postulated release of radioactive material to within limits. The test interval associated with Type A ILRTs is not a precursor of any accident previously evaluated. Type A ILRTs provide assurance that the TMI, Unit 1 containment will not exceed allowable leakage rate values specified in the TS and will continue to perform its design function following an accident. The risk assessment of the proposed change has concluded that there is an insignificant increase in postulated total population dose rate and an insignificant increase in the postulated conditional containment failure probability. Additionally, containment inspections have also been performed which demonstrate the continued structural integrity of the primary containment.

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

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

Response: No

The proposed change for a one-time extension of the Type A ILRT for TMI, Unit 1 will not affect the control parameters governing unit operation or the response of plant equipment to transient and accident conditions. The proposed change does not introduce any new equipment, modes of system operation or failure mechanisms.

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

Does the proposed amendment involve a significant reduction in a margin of safety?

Response: No

The integrity of the containment penetrations and isolation valves is verified through Type B and Type C local leak rate tests (LLRTs) and the overall leak tight integrity of the containment is verified by a Type A ILRT, as required by 10 CFR 50, Appendix J, "Primary Reactor Containment Leakage Testing for Water-Cooled Power Reactors." These tests are performed to verify the essentially leak tight characteristics of the containment at the design basis accident pressure. The proposed change for a one-time extension of the Type A ILRT does not affect the method for Type A, B or C testing or the test acceptance criteria.

AmerGen has conducted a risk assessment to determine the impact of a change to the TMI, Unit 1 Type A ILRT schedule from a baseline ILRT frequency of three times in 10 years to once in 15 years plus 15 months for the risk measures of Large Early Release Frequency (i.e., LERF), Population Dose, and Conditional Containment Failure Probability (i.e., CCFP). This assessment indicated that the proposed TMI, Unit 1 ILRT interval extension has a small change in risk to the public and is an acceptable plant change from a risk perspective.

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

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

5.2 REGULATORY REQUIREMENTS/CRITERIA

10 CFR 50.36, "Technical Specifications," provides the regulatory requirements for the content required in a plant's Technical Specifications (TS). 10 CFR 50.36(c)(5), "Administrative controls," requires provisions relating to organization and management, procedures, recordkeeping, review and audit, and reporting necessary to assure operation of the facility in a safe manner be included in a plant's TS.

Additionally, 10 CFR 50, Appendix J, Option B, Section V.B.3, "Implementation," specifies that the regulatory guide or other implementing documents used to develop a performance-based leakage testing program must be included, by general reference, in the plant's TS. Additionally, deviations from guidelines endorsed in a regulatory guide are to be submitted as a revision to the plant's TS.

The proposed change will revise TS 6.8.5 to reflect a one-time extension to the Three Mile Island, Unit 1 Type A ILRT as currently specified in the Technical Specifications. The one-time extension deviates from the guidelines contained in Regulatory Guide (RG) 1.163. The proposed change is consistent with the requirements of 10 CFR 50.36(c)(5) and 10 CFR 50, Appendix J, Option B, Section V.B.3.

Additionally, in accordance with 10 CFR 50, Appendix J, Option B, Section V. B, the proposed change to the TMI, Unit 1 TS does not require a supporting request for an exemption to Option B of Appendix J, in accordance with 10 CFR 50.12, "Specific exemptions."

6.0 ENVIRONMENTAL CONSIDERATION

A review has determined that the proposed amendment would change a requirement with respect to installation or use of a facility component located within the restricted area, as defined in 10 CFR 20, or would change an inspection or surveillance requirement. However, the proposed amendment does not involve: (i) a significant hazards consideration, (ii) a significant change in the types or significant increase in the amounts of any effluent that may be released offsite, or (iii) a significant increase in individual or cumulative occupational radiation exposure. Accordingly, the proposed amendment meets the eligibility criterion for categorical exclusion set forth in 10 CFR 51.22(c)(9). Therefore, pursuant to 10 CFR 51.22(b), no environmental impact statement or environmental assessment need be prepared in connection with the proposed amendment.

7.0 PRECEDENT

The proposed amendment incorporates into the TMI, Unit 1 TS a change that is similar to changes (i.e., greater than 15 years) approved by the NRC for St. Lucie Station on December 23, 2005, River Bend Station on February 9, 2006, and Seabrook Station on March 24, 2006.

8.0 REFERENCES

(1) Nuclear Energy Institute (NEI) 94-01, Revision 0, "Industry Guideline for Implementing Performance-Based Option of 10 CFR 50, Appendix J," dated July 26, 1995

(2) AmerGen letter 5928-02-21096 dated September 30, 2002, "License Amendment Request No. 318 – Integrated Leak Rate Test Deferral"

(3) AmerGen letter 5928-03-20050 dated March 19, 2003, "License Amendment Request No. 318 – Integrated Leak Rate Test Deferral Response to Request for Additional Information"

(4) AmerGen letter 5928-03-20158, "License Amendment Request No. 318 – Integrated Leak Rate Test Deferral Submittal of Camera Ready Page"

(5) USNRC letter dated August 14, 2003, "Three Mile Island Nuclear Station, Unit 1 (TMI-1) RE: Deferral of Containment Integrated Leakage Rate Test" (TAC NO. MB6487)

(6) NUREG-1493, "Performance-Based Containment Leak-Test Program," dated July 1995

(7) "Risk Impact Assessment of Extended Integrated Leak Rate Testing Intervals," EPRI TR-1009325, Revision 1, dated December 2005

(8) Regulatory Guide 1.174, "An Approach for Using Probabilistic Risk Assessment in Risk-Informed Decisions On Plant-Specific Changes to the Licensing Basis," Rev. 1, November 2002

(9) "Interim Guidance for Performing Risk Impact Assessments In Support of One-Time Extensions for Containment Integrated Leakage Rate Test Surveillance Intervals," Developed for NEI by EPRI, Jack Haugh (EPRI Area Manager, Nuclear Safety & Analysis), John M. Gisclon (EPRI Consultant), William Parkinson and Ken Canavan (Data Systems and Solutions), dated November 2001

ATTACHMENT 2

MARKUP OF PROPOSED TECHNICAL SPECIFICATION PAGE CHANGE

Revised TS Pages

6-11c

6.8.5 Reactor Building Leakage Rate Testing Program

The Reactor Building Leakage Rate Testing Program shall be established, implemented, and maintained as follows:

A program shall be established to implement the leakage rate testing of the Reactor Building 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, as modified by the following exception to NEI 94-01, Rev. 0, "Industry Guideline for Implementing Performance-Based Option of 10 CFR Part 50, Appendix J":

a. Section 9.2.3: The first Type A test performed after the September 1993 Type A test shall be performed no later than September 2008.

The peak calculated Reactor Building internal pressure for the design basis loss of coolant accident, P_{ac}, is 50.6 psig.

The maximum allowable Reactor Building leakage rate, L_a, shall be 0.1 weight percent of containment atmosphere per 24 hours at P_{ac}.

Reactor Building leakage rate acceptance criteria is $\leq 1.0\ L_a$. During the first plant startup following each test performed in accordance with this program, the leakage rate acceptance criteria are $\leq 0.60\ L_a$ for the Type B and Type C tests and $\leq 0.75\ L_a$ for the Type A tests.

prior to startup from the T1R18 refueling outage

ATTACHMENT 3

RETYPED PAGE FOR TECHNICAL SPECIFICATION CHANGE

<u>Retyped TS Pages</u>

6-11c

6.8.5 Reactor Building Leakage Rate Testing Program

The Reactor Building Leakage Rate Testing Program shall be established, implemented, and maintained as follows:

A program shall be established to implement the leakage rate testing of the Reactor Building 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, as modified by the following exception to NEI 94-01, Rev. 0, "Industry Guideline for Implementing Performance-Based Option of 10 CFR Part 50, Appendix J":

a. Section 9.2.3: The first Type A test performed after the September 1993 Type A test shall be performed prior to startup from the T1R18 refueling outage.

The peak calculated Reactor Building internal pressure for the design basis loss of coolant accident, P_{ac}, is 50.6 psig.

The maximum allowable Reactor Building leakage rate, L_a, shall be 0.1 weight percent of containment atmosphere per 24 hours at P_{ac}.

Reactor Building leakage rate acceptance criteria is $\leq 1.0\ L_a$. During the first plant startup following each test performed in accordance with this program, the leakage rate acceptance criteria are $\leq 0.60\ L_a$ for the Type B and Type C tests and $\leq 0.75\ L_a$ for the Type A tests.

ATTACHMENT 4

Risk Assessment for TMI Unit 1 to Support ILRT (Type A) Interval Extension Request

RM DOCUMENTATION NO.	TMI-2006-003	REV: 0	PAGE NO. 1

STATION: THREE MILE ISLAND

UNIT(S) AFFECTED: 1

TITLE: Risk Assessment for TMI Unit 1 To Support ILRT (Type A) Interval Extension Request

SUMMARY (Include UREs incorporated): The purpose of this analysis is to provide an assessment of the risk associated with implementing a one-time extension of the Three Mile Island Unit 1 (TMI-1) containment Type A integrated leak rate test (ILRT) interval from 15 years to 16.25 years (an additional 15-month extension).

Internal RM Documentation

Electronic Calculation Data Files:
(Program Name, Version, File Name extension/size/date/hour/min)

Prepared by: Robert J. Wolfgang/Don E. Vanover / Robert J. Wolfgang /DE Vanover / 6/12/06
Print Sign Date

Reviewed by: Leo B. Shanley / [signature] / 6/12/06
Print Sign Date

Method of Review: [X] Detailed [ ] Alternate

This RM documentation supersedes: N/A **in its entirety.**

Approved by: JEFF R GABOR / [signature] / 6/12/06
Print Sign Date

External RM Documentation

Reviewed by: N/A / N/A /
Print Sign Date

Approved by: N/A / N/A /
Print Sign Date

Do any ASSUMPTIONS / ENGINEERING JUDGEMENTS require later verification? **[ ] Yes [X] No**

Tracked By: AT#, URE# etc.)

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FOREWORD

This analysis is to support an extension of the Three Mile Island (TMI) Unit 1 ILRT interval from the currently approved once-in-15 year interval to once-in-16.25 years (15 years and 15 months).

A similar analysis was submitted for a one-time extension for the TMI-1 ILRT interval in 2003 to increase the interval from once-in-10 years to once-in-15 years.

The current analysis is based on the 2004 Revision 1 PRA model which is the current TMI-1 model of record, which is an update of the one used for the 2003 submittal.

The enclosed analysis demonstrates the one-time ILRT interval extension to once-in-16.25 years results in a small change in risk when comparing the results to the standard baseline 3-in-10 year testing frequency.

Section 1
INTRODUCTION

1.1 PURPOSE AND IDENTIFICATION OF SSC

The purpose of this analysis is to provide an assessment of the risk associated with implementing a one-time extension of the Three Mile Island Unit 1 (TMI-1) containment Type A integrated leak rate test (ILRT) interval from 15 years to 16.25 years (an additional 15-month extension). The applicable safety system, structure, or component (SSC) is the reactor building, also referred to as the containment vessel. A similar risk assessment was performed previously to support a one-time extension of the ILRT interval from 10 to 15 years [23]. 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], 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], and the methodology outlined in a response to a Request for Additional Information for a previous TMI-1 ILRT submittal [32], which used an approach similar to what was done in a study for Calvert Cliffs to estimate the likelihood and risk implications of corrosion-induced leakage of steel liners going undetected during the extended test interval. The format of this document is consistent with the intent of the Risk Impact Assessment Template for evaluating extended integrated leak rate testing intervals provided in the December 2005 EPRI final Report [33].

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 that for a representative PWR plant (i.e., Surry) that containment isolation failures contribute less than 0.1 percent to the latent risks from reactor accidents. Consequently, 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 TMI-1 specific models and available data.

Earlier ILRT frequency extension submittals have used the EPRI TR-104285 methodology to perform the risk assessment. In November and December 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 TMI-1 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, this analysis uses the following risk metrics to characterize the change in risk associated with the one time ILRT extension:

- 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. NRC

Regulatory Guide 1.174, "An Approach for Using PRA in Risk-Informed Decisions on Plant-Specific Changes to the Licensing Basis", provides NRC recommendations for using risk information in support of applications requesting changes to the license basis of the plant.

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. "Small" changes are defined as increases in CDF less than 10^{-5} per reactor year and increases in LERF less than 10^{-6} 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 conditional containment failure probability 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

This section provides the following methodology related items:

- Brief summary of available resource documents to support the methodology
- NEI Interim Guidance for the analysis approach to be used
- General assumptions used in the evaluation
- Plant-specific inputs

2.1 General Resources Available

This section summarizes the general resources available as input. 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]
8) NEI Interim Guidance [3,21]

The first study is applicable because it provides one basis for the threshold that could be used in the Level 2 PRA for the size of containment leakage that is considered significant and 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. Finally, the eighth study includes the NEI recommended methodology for evaluating the risk associated with obtaining a one-time extension of the ILRT interval.

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 PWR events in 740 reactor years and conservatively assuming a one-year duration for each event.

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 (shutdown CDF reduced by 1E-8/yr to 1E-7/yr) is realized from extending the test interval from 3 per 10 years to 1 per 10 years.

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. . ."*

<u>NEI Interim Guidance [3,21]</u>

NEI "Interim Guidance for Performing Risk Impact Assessments in Support of One-Time Extensions of Containment Integrated Leakage Rate Test Surveillance Intervals" [3] has been developed to provide utilities with revised guidance regarding licensing submittals. Additional information from NEI on the "Interim Guidance" was supplied in Reference [21].

A nine step process is defined which includes changes in the following areas of the previous EPRI guidance:

- Impact of extending surveillance intervals on dose

- Method used to calculate the frequencies of leakages detectable only by ILRTs
- Provisions for using NUREG-1150 [14] dose calculations to support the population dose determination.

This NEI Guidance is used in the TMI-1 ILRT interval extension risk assessment.

2.2 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 categories 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 and 3 where category 3 is subdivided into categories 3a and 3b for "small" and "large" isolation failures, respectively.
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 forms 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.

2.3 ASSUMPTIONS

The following ground rules are used in the analysis:

- The TMI-1 internal events Level 1 and Level 2 PRAs are reflective of the current as-built plant and provide reasonable representative risk spectrum results for use in this analysis. External event risk results from the TMI-1 IPEEE are investigated as a sensitivity discussion.
- The base Level 3 results are obtained from a separate analysis for TMI-1, based on a generic B&W Owners Group Level 3 PRA methodology [22].
- The use of year 2000 population data was considered adequate for this 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]
- The reduction in ILRT frequency does not impact the reliability of containment isolation valves to close in response to a containment isolation signal.

2.4 PLANT-SPECIFIC INPUTS

The TMI-1 specific information used to perform this ILRT interval extension risk assessment includes the following:

- TMI-1 Internal Events Level 1 PRA [29]
- TMI-1 Containment Safeguards Event Tree and Associated System Models [26, 27, 28, and 30]
- TMI-1 Internal Events Level 2 PRA [25]
- TMI-1 Internal Events Level 3 PRA [19]
- TMI-1 IPEEE [31]
- Past TMI-1 ILRT results to demonstrate adequacy of the administrative and hardware issues.

2.4.1 TMI-1 Internal Events Level 1 PRA

The TMI-1 Level 1 PRA used as input to this analysis is characteristic of the as-built, as-operated plant. The current Level 1 PRA model was developed in CAFTA. The total internal events core damage frequency (CDF) used in this analysis was 3.19E-5/yr. Table 2-2 summarizes the TMI-1 Level 1 PRA core damage frequency results by plant damage state (PDS). The TMI-1 internal events Level 1 PRA model [29] was used in conjunction with a Level 1 to Level 2 "bridge" event tree and associated containment safeguards system fault tree models to bin each core damage cutset into a defined PDS [26, 27, 28, and 30]. The details that define how each core damage sequence was binned to a specific PDS is found in Reference [30].

As a result of implementing an updated Level 1 PRA model with a new Level 1 to Level 2 interface, new PDSs resulted that had not been present before with the previous Level 2 model [25]. Because of this, there was no direct relationship between the new PDSs and existing release categories; therefore, new PDS frequencies were binned with existing PDS results, with priority on maintaining the same (or more conservative) containment safeguards status. Table 2-1 shows the mapping scheme and resulting values for the established PDS frequency categories resulting from this new ILRT extension analysis. This process was meant to preserve the existing PDS fraction that was assigned to each release category, which was then used in calculating the results that appear in Table 2-3. This same fractional assignment of PDSs to release categories was also used in a previous TMI-1 ILRT extension analysis [23].

2.4.2 TMI-1 Internal Events Level 2 PRA

Table 2-3 summarizes the pertinent TMI-1 internal events Level 2 PRA results in terms of release category as a function of plant damage state using the TMI-1 Internal Events Level 2 PRA model [25]. As discussed in the notes to Table 2-3, release categories RC901 through RC904 refer to severe accidents with no containment failure. The total frequency of accidents in which the containment remains intact (i.e., containment leakage within Technical Specifications) is 1.28E-5/yr.

2.4.3 TMI-1 Internal Events Level 3 PRA

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

Although TMI-1 does not maintain a Level 3 PRA, a generic Level 3 PRA applicable to Babcock and Wilcox (B&W) PWR plants was performed by the B&W Owners Group (B&WOG) [19]. The generic Level 3 PRA provided by the B&WOG in Reference [19] was enhanced in support of this ILRT risk assessment to incorporate the following:

- TMI-1 specific meteorological data
- TMI-1 specific population data (year 2000)
- Core radioisotope inventory parameters representative of the TMI-1 24-month fuel cycle.
- TMI-1 specific fission product release source term information for many of the release categories
- Revised baseline and sensitivity MACCS2 [17] consequence calculations

This supporting calculation is contained in TMI Calculation No. C-1101-900-E-220-178 [22]. The results from that analysis are used as direct input in this risk assessment to assign 50-mile radius population doses (refer to Section 3.3). Consequently, the assumptions utilized in C-1101-900-E-220-178 are implicitly included with this analysis. The TMI-1 specific doses by release category are summarized in Table 2-4 (this table also includes the release category frequency and dose rate).

Table 2-1
TMI-1 ASSIGNMENT OF PLANT DAMAGE STATE BINS

Plant Damage State Category	Assumed Additional PDS Contribution	New Core Damage Frequency (1/yr)	Sum Total of CDF for PDS Category (1/yr)
PDS-01A		9.35E-08	**9.35E-08**
PDS-01C		1.26E-11	**1.21E-09**
	PDS-01D	*1.20E-09*	
PDS-02B		6.85E-08	**6.85E-08**
PDS-03A		2.43E-09	**2.43E-09**
PDS-04A		1.02E-08	**1.02E-08**
PDS-04B		5.35E-09	**1.05E-08**
	PDS-07B	*5.19E-09*	
PDS-04C		3.93E-10	**2.01E-09**
	PDS-02D	*1.62E-09*	
PDS-04F		1.62E-09	**1.85E-09**
	PDS-01F	*2.33E-10*	
PDS-05A			**1.62E-07**
	PDS-02A	*1.62E-07*	
PDS-05B		7.19E-09	**1.17E-08**
	PDS-06B	*3.84E-09*	
	PDS-03B	*7.11E-10*	
PDS-05C			**1.56E-08**
	PDS-02C	*1.02E-11*	
	PDS-03C	*2.26E-09*	
	PDS-06C	*1.33E-08*	
	PDS-09C	*1.26E-11*	
PDS-05F			**9.51E-07**
	PDS-02E	*9.34E-10*	
	PDS-05E	*8.33E-11*	
	PDS-08F	*1.98E-07*	
	PDS-09F	*7.52E-07*	
	PDS-03F	*1.13E-11*	
PDS-06A		1.42E-08	**1.42E-08**
PDS-07A		5.43E-06	**5.43E-06**
PDS-07C		3.07E-06	**3.07E-06**
PDS-07D		1.07E-07	**1.24E-07**
	PDS-04D	*2.89E-10*	
	PDS-12D	*1.63E-08*	
PDS-07E			**3.27E-10**
	PDS-04E	*3.27E-10*	
PDS-07F		2.55E-06	**2.63E-06**

Table 2-1
TMI-1 ASSIGNMENT OF PLANT DAMAGE STATE BINS

Plant Damage State Category	Assumed Additional PDS Contribution	New Core Damage Frequency (1/yr)	Sum Total of CDF for PDS Category (1/yr)
	PDS-12F	*7.62E-08*	
PDS-07L		3.23E-08	**5.25E-08**
	PDS-07G	*4.51E-10*	
	PDS-07I	*5.06E-09*	
	PDS-07J	*2.50E-10*	
	PDS-08G	*1.38E-09*	
	PDS-08H	*9.14E-10*	
	PDS-08I	*4.10E-11*	
	PDS-08L	*2.95E-09*	
	PDS-09L	*8.55E-09*	
	PDS-12G	*3.28E-11*	
	PDS-12L	*5.12E-10*	
	PDS-15L	*4.68E-11*	
PDS-08A		7.59E-06	**8.20E-06**
	PDS-13A	*6.12E-07*	
PDS-08B		2.80E-06	**2.89E-06**
	PDS-12B	*3.51E-10*	
	PDS-13B	*6.44E-08*	
	PDS-14B	*2.71E-08*	
PDS-08C		4.45E-06	**4.45E-06**
PDS-08D		7.76E-08	**8.44E-08**
	PDS-13D	*6.67E-09*	
	PDS-14D	*1.75E-10*	
PDS-08E		2.85E-08	**3.00E-08**
	PDS-02E	*9.34E-10*	
	PDS-06E	*6.77E-11*	
	PDS-13E	*4.86E-10*	
	PDS-14E	*2.30E-11*	
PDS-10A		1.08E-09	**1.45E-06**
	PDS-09A	*2.93E-10*	
	PDS-12A	*1.35E-06*	
	PDS-14A	*9.45E-08*	
PDS-10C			**1.04E-07**
	PDS-14C	*1.04E-07*	
PDS-12C		1.40E-08	**1.40E-08**
PDS-15A		6.39E-09	**1.30E-07**
	PDS-15C	*1.78E-09*	
	PDS-15D	*2.63E-09*	
	PDS-15F	*1.88E-08*	
	PDS-16A	*9.98E-08*	

Table 2-1
TMI-1 ASSIGNMENT OF PLANT DAMAGE STATE BINS

Plant Damage State Category	Assumed Additional PDS Contribution	New Core Damage Frequency (1/yr)	Sum Total of CDF for PDS Category (1/yr)
	PDS-16D	*1.03E-09*	
PDS-18C		1.66E-06	**1.74E-06**
	PDS-16C	*7.62E-08*	
PDS-18F		2.10E-08	**2.22E-08**
	PDS-16F	*1.15E-09*	
PDS-19O			**1.83E-07**
	PDS-07R	*1.81E-10*	
	PDS-19A	*1.80E-07*	
	PDS-19B	*5.94E-11*	
	PDS-19C	*6.17E-11*	
	PDS-19D	*2.36E-09*	

Table 2-2

TMI-1 CORE DAMAGE FREQUENCY BY PLANT DAMAGE STATE

Plant Damage State	Core Damage Frequency (1/yr)	% of CDF
PDS-01A	9.35E-08	0.3%
PDS-01C	1.21E-09	0.0%
PDS-02B	6.85E-08	0.2%
PDS-03A	2.43E-09	0.0%
PDS-04A	1.02E-08	0.0%
PDS-04B	1.05E-08	0.0%
PDS-04C	2.01E-09	0.0%
PDS-04F	1.85E-09	0.0%
PDS-04L	0.00E+00	0.0%
PDS-05A	1.62E-07	0.5%
PDS-05B	1.17E-08	0.0%
PDS-05C	1.56E-08	0.0%
PDS-05F	9.51E-07	3.0%
PDS-06A	1.42E-08	0.0%
PDS-07A	5.43E-06	17.0%
PDS-07C	3.07E-06	9.6%
PDS-07D	1.24E-07	0.4%
PDS-07E	3.27E-10	0.0%
PDS-07F	2.63E-06	8.2%
PDS-07L	5.25E-08	0.2%
PDS-08A	8.20E-06	25.7%
PDS-08B	2.89E-06	9.1%
PDS-08C	4.45E-06	13.9%
PDS-08D	8.44E-08	0.3%
PDS-08E	3.00E-08	0.1%
PDS-10A	1.45E-06	4.5%
PDS-10C	1.04E-07	0.3%
PDS-12C	1.40E-08	0.0%
PDS-15A	1.30E-07	0.4%
PDS-18B	0.00E+00	0.0%
PDS-18C	1.74E-06	5.4%
PDS-18E	0.00E+00	0.0%
PDS-18F	2.22E-08	0.1%
PDS-19O	1.83E-07	0.6%
Total	3.19E-05	100.0%

Notes to Table 2-2:

The TMI-1 Plant Damage States (PDSs) are defined using a two-term nomenclature. The first term is the Core Melt Bin (a numeric designator) and the second term is the Containment Safeguards and Isolation State (an alphabetical designator). These designators are summarized below:

First Term (Core Melt Bin):

1	Large LOCA, injection failure
2	Large LOCA, early recirculation failure
3	Large LOCA, late recirculation failure
4	Medium LOCA, injection failure
5	Medium LOCA, early recirculation failure
6	Medium LOCA, late recirculation failure
7	Small LOCA, injection failure, steam generators available
8	Small LOCA, recirculation failure, steam generators available
9	Small LOCA, injection failure, steam generators unavailable
10	Small LOCA, early recirculation failure, steam generators unavailable
11	Small LOCA, late recirculation failure, steam generators unavailable
12	Cycling relief valve, injection failure
13	Cycling relief valve, early recirculation failure
14	Cycling relief valve, late recirculation failure
15	Steam generator tube rupture, injection failure, steam generators unavailable
16	Steam generator tube rupture, early recirculation failure, steam generators unavailable
17	Steam generator tube rupture, late recirculation failure, steam generators unavailable
18	Steam generator tube rupture, steam generators available
19	Interfacing-systems LOCA

Second Term (Containment Safeguards and Isolation State):

A	All safeguards available, containment isolated
B	Fans available, sprays available in injection mode; sprays unavailable in recirculation mode, containment isolated
C	Fans available; sprays unavailable in injection and recirculation modes, containment isolated
D	Sprays available in injection and recirculation modes; fans unavailable, containment isolated
E	Sprays in injection mode available; fans unavailable, sprays unavailable in recirculation mode, containment isolated
F	No safeguards available, containment isolated
G	All safeguards available, small isolation failure
H	Fans available, sprays available in injection mode; sprays unavailable in recirculation mode, small isolation failure
I	Fans available; sprays unavailable in injection and recirculation modes, small isolation failure
J	Sprays available in injection and recirculation modes; fans unavailable, small isolation failure
K	Sprays in injection mode available; fans unavailable, sprays unavailable in recirculation mode, small isolation failure
L	No safeguards available, small isolation failure
M	All safeguards available, large isolation failure
N	Fans available, sprays available in injection mode; sprays unavailable in recirculation mode, large isolation failure
O	Fans available; sprays unavailable in injection and recirculation modes, large isolation failure
P	Sprays available in injection and recirculation modes; fans unavailable, large isolation failure
Q	Sprays in injection mode available; fans unavailable, sprays unavailable in recirculation mode, large isolation failure
R	No safeguards available, large isolation failure

Table 2-3
TMI-1 RADIONUCLIDE RELEASE CATEGORY FREQUENCY BY PLANT DAMAGE STATE (page 1 of 2)

	RC101	RC102	RC103	RC104	RC201	RC202	RC203	RC204	RC301	RC302	RC303	RC304	RC305	RC306	RC401	RC402	RC403	RC404	RC405	RC406	RC407
PDS-01A	0.00E+0	0.00E+0	0.00E+0	0.00E+0	0.00E+0	0.00E+0	0.00E+0	0.00E+0	0.00E+0	0.00E+0	0.00E+0	0.00E+0	0.00E+0	0.00E+0	0.00E+0	0.00E+0	0.00E+0	0.00E+0	0.00E+0	0.00E+0	0.00E+0
PDS-01C	0.00E+0	0.00E+0	0.00E+0	0.00E+0	0.00E+0	0.00E+0	0.00E+0	0.00E+0	0.00E+0	0.00E+0	0.00E+0	0.00E+0	0.00E+0	0.00E+0	0.00E+0	0.00E+0	0.00E+0	0.00E+0	0.00E+0	0.00E+0	0.00E+0
PDS-02B	0.00E+0	0.00E+0	0.00E+0	0.00E+0	0.00E+0	0.00E+0	0.00E+0	0.00E+0	0.00E+0	0.00E+0	0.00E+0	0.00E+0	0.00E+0	0.00E+0	0.00E+0	0.00E+0	0.00E+0	0.00E+0	0.00E+0	0.00E+0	0.00E+0
PDS-03A	0.00E+0	0.00E+0	0.00E+0	0.00E+0	0.00E+0	0.00E+0	0.00E+0	0.00E+0	0.00E+0	0.00E+0	0.00E+0	0.00E+0	0.00E+0	0.00E+0	0.00E+0	0.00E+0	0.00E+0	0.00E+0	0.00E+0	0.00E+0	0.00E+0
PDS-04A	0.00E+0	0.00E+0	0.00E+0	0.00E+0	0.00E+0	0.00E+0	0.00E+0	0.00E+0	0.00E+0	0.00E+0	0.00E+0	0.00E+0	0.00E+0	0.00E+0	0.00E+0	0.00E+0	0.00E+0	0.00E+0	0.00E+0	0.00E+0	0.00E+0
PDS-04B	0.00E+0	0.00E+0	0.00E+0	0.00E+0	0.00E+0	0.00E+0	0.00E+0	0.00E+0	0.00E+0	0.00E+0	0.00E+0	0.00E+0	0.00E+0	0.00E+0	0.00E+0	0.00E+0	0.00E+0	0.00E+0	0.00E+0	0.00E+0	0.00E+0
PDS-04C	0.00E+0	0.00E+0	0.00E+0	0.00E+0	0.00E+0	0.00E+0	0.00E+0	0.00E+0	0.00E+0	0.00E+0	0.00E+0	0.00E+0	0.00E+0	0.00E+0	0.00E+0	0.00E+0	0.00E+0	0.00E+0	0.00E+0	0.00E+0	0.00E+0
PDS-04F	0.00E+0	0.00E+0	0.00E+0	0.00E+0	0.00E+0	0.00E+0	0.00E+0	0.00E+0	0.00E+0	0.00E+0	0.00E+0	0.00E+0	0.00E+0	0.00E+0	0.00E+0	0.00E+0	0.00E+0	0.00E+0	0.00E+0	0.00E+0	0.00E+0
PDS-04L	0.00E+0	0.00E+0	0.00E+0	0.00E+0	0.00E+0	0.00E+0	0.00E+0	0.00E+0	0.00E+0	0.00E+0	0.00E+0	0.00E+0	0.00E+0	0.00E+0	0.00E+0	0.00E+0	0.00E+0	0.00E+0	0.00E+0	0.00E+0	0.00E+0
PDS-05A	0.00E+0	0.00E+0	0.00E+0	0.00E+0	0.00E+0	0.00E+0	0.00E+0	0.00E+0	0.00E+0	0.00E+0	0.00E+0	0.00E+0	0.00E+0	0.00E+0	0.00E+0	0.00E+0	0.00E+0	0.00E+0	0.00E+0	0.00E+0	0.00E+0
PDS-05B	0.00E+0	0.00E+0	0.00E+0	0.00E+0	0.00E+0	0.00E+0	0.00E+0	0.00E+0	0.00E+0	0.00E+0	0.00E+0	0.00E+0	0.00E+0	0.00E+0	0.00E+0	0.00E+0	0.00E+0	0.00E+0	0.00E+0	0.00E+0	0.00E+0
PDS-05C	0.00E+0	0.00E+0	0.00E+0	0.00E+0	0.00E+0	0.00E+0	0.00E+0	0.00E+0	0.00E+0	0.00E+0	0.00E+0	0.00E+0	0.00E+0	0.00E+0	0.00E+0	0.00E+0	0.00E+0	0.00E+0	0.00E+0	0.00E+0	0.00E+0
PDS-05F	0.00E+0	0.00E+0	0.00E+0	0.00E+0	0.00E+0	0.00E+0	0.00E+0	0.00E+0	0.00E+0	0.00E+0	0.00E+0	0.00E+0	0.00E+0	0.00E+0	0.00E+0	0.00E+0	0.00E+0	0.00E+0	0.00E+0	0.00E+0	0.00E+0
PDS-06A	0.00E+0	0.00E+0	0.00E+0	0.00E+0	0.00E+0	0.00E+0	0.00E+0	0.00E+0	0.00E+0	0.00E+0	0.00E+0	0.00E+0	0.00E+0	0.00E+0	0.00E+0	0.00E+0	0.00E+0	0.00E+0	0.00E+0	0.00E+0	0.00E+0
PDS-07A	0.00E+0	0.00E+0	0.00E+0	0.00E+0	0.00E+0	0.00E+0	0.00E+0	0.00E+0	0.00E+0	0.00E+0	0.00E+0	0.00E+0	0.00E+0	0.00E+0	0.00E+0	0.00E+0	0.00E+0	0.00E+0	0.00E+0	0.00E+0	0.00E+0
PDS-07C	0.00E+0	0.00E+0	0.00E+0	0.00E+0	0.00E+0	0.00E+0	0.00E+0	0.00E+0	0.00E+0	0.00E+0	0.00E+0	0.00E+0	0.00E+0	0.00E+0	0.00E+0	0.00E+0	0.00E+0	0.00E+0	0.00E+0	0.00E+0	0.00E+0
PDS-07D	0.00E+0	0.00E+0	0.00E+0	0.00E+0	0.00E+0	0.00E+0	0.00E+0	0.00E+0	0.00E+0	0.00E+0	0.00E+0	0.00E+0	0.00E+0	0.00E+0	0.00E+0	0.00E+0	0.00E+0	0.00E+0	0.00E+0	0.00E+0	0.00E+0
PDS-07E	0.00E+0	0.00E+0	0.00E+0	0.00E+0	0.00E+0	0.00E+0	0.00E+0	0.00E+0	0.00E+0	0.00E+0	0.00E+0	0.00E+0	0.00E+0	0.00E+0	0.00E+0	0.00E+0	0.00E+0	0.00E+0	0.00E+0	0.00E+0	0.00E+0
PDS-07F	0.00E+0	0.00E+0	0.00E+0	0.00E+0	0.00E+0	0.00E+0	0.00E+0	0.00E+0	0.00E+0	0.00E+0	0.00E+0	0.00E+0	0.00E+0	0.00E+0	0.00E+0	0.00E+0	0.00E+0	0.00E+0	0.00E+0	0.00E+0	0.00E+0
PDS-07L	0.00E+0	0.00E+0	0.00E+0	0.00E+0	0.00E+0	0.00E+0	0.00E+0	0.00E+0	0.00E+0	0.00E+0	0.00E+0	0.00E+0	0.00E+0	0.00E+0	0.00E+0	0.00E+0	4.67E-8	5.19E-9	0.00E+0	0.00E+0	0.00E+0
PDS-08A	0.00E+0	0.00E+0	0.00E+0	0.00E+0	0.00E+0	0.00E+0	0.00E+0	0.00E+0	0.00E+0	0.00E+0	0.00E+0	0.00E+0	0.00E+0	0.00E+0	0.00E+0	0.00E+0	0.00E+0	0.00E+0	0.00E+0	0.00E+0	0.00E+0
PDS-08B	0.00E+0	0.00E+0	0.00E+0	0.00E+0	0.00E+0	0.00E+0	0.00E+0	0.00E+0	0.00E+0	0.00E+0	0.00E+0	0.00E+0	0.00E+0	0.00E+0	0.00E+0	0.00E+0	0.00E+0	0.00E+0	0.00E+0	0.00E+0	0.00E+0
PDS-08C	0.00E+0	0.00E+0	0.00E+0	0.00E+0	0.00E+0	0.00E+0	0.00E+0	0.00E+0	0.00E+0	0.00E+0	0.00E+0	0.00E+0	0.00E+0	0.00E+0	0.00E+0	0.00E+0	0.00E+0	0.00E+0	0.00E+0	0.00E+0	0.00E+0
PDS-08D	0.00E+0	0.00E+0	0.00E+0	0.00E+0	0.00E+0	0.00E+0	0.00E+0	0.00E+0	0.00E+0	0.00E+0	0.00E+0	0.00E+0	0.00E+0	0.00E+0	0.00E+0	0.00E+0	0.00E+0	0.00E+0	0.00E+0	0.00E+0	0.00E+0
PDS-08E	0.00E+0	0.00E+0	0.00E+0	0.00E+0	0.00E+0	0.00E+0	0.00E+0	0.00E+0	0.00E+0	0.00E+0	0.00E+0	0.00E+0	0.00E+0	0.00E+0	0.00E+0	0.00E+0	0.00E+0	0.00E+0	0.00E+0	0.00E+0	0.00E+0
PDS-10A	0.00E+0	1.20E-6	0.00E+0	6.32E-8	0.00E+0	0.00E+0	0.00E+0	0.00E+0	0.00E+0	0.00E+0	0.00E+0	0.00E+0	0.00E+0	0.00E+0	0.00E+0	0.00E+0	0.00E+0	0.00E+0	0.00E+0	0.00E+0	0.00E+0
PDS-10C	0.00E+0	0.00E+0	0.00E+0	9.08E-8	0.00E+0	0.00E+0	0.00E+0	0.00E+0	0.00E+0	0.00E+0	0.00E+0	0.00E+0	0.00E+0	0.00E+0	0.00E+0	0.00E+0	0.00E+0	0.00E+0	0.00E+0	0.00E+0	0.00E+0
PDS-12C	0.00E+0	0.00E+0	0.00E+0	1.10E-8	0.00E+0	0.00E+0	0.00E+0	0.00E+0	0.00E+0	0.00E+0	0.00E+0	0.00E+0	0.00E+0	0.00E+0	0.00E+0	0.00E+0	0.00E+0	0.00E+0	0.00E+0	0.00E+0	0.00E+0
PDS-15A	0.00E+0	1.24E-7	0.00E+0	6.52E-9	0.00E+0	0.00E+0	0.00E+0	0.00E+0	0.00E+0	0.00E+0	0.00E+0	0.00E+0	0.00E+0	0.00E+0	0.00E+0	0.00E+0	0.00E+0	0.00E+0	0.00E+0	0.00E+0	0.00E+0
PDS-18B	0.00E+0	0.00E+0	0.00E+0	0.00E+0	0.00E+0	0.00E+0	0.00E+0	0.00E+0	0.00E+0	0.00E+0	0.00E+0	0.00E+0	0.00E+0	0.00E+0	0.00E+0	0.00E+0	0.00E+0	0.00E+0	0.00E+0	0.00E+0	0.00E+0
PDS-18C	0.00E+0	0.00E+0	8.68E-7	8.68E-7	0.00E+0	0.00E+0	0.00E+0	0.00E+0	0.00E+0	0.00E+0	0.00E+0	0.00E+0	0.00E+0	0.00E+0	0.00E+0	0.00E+0	0.00E+0	0.00E+0	0.00E+0	0.00E+0	0.00E+0
PDS-18E	0.00E+0	0.00E+0	0.00E+0	0.00E+0	0.00E+0	0.00E+0	0.00E+0	0.00E+0	0.00E+0	0.00E+0	0.00E+0	0.00E+0	0.00E+0	0.00E+0	0.00E+0	0.00E+0	0.00E+0	0.00E+0	0.00E+0	0.00E+0	0.00E+0
PDS-18F	0.00E+0	0.00E+0	1.11E-8	1.11E-8	0.00E+0	0.00E+0	0.00E+0	0.00E+0	0.00E+0	0.00E+0	0.00E+0	0.00E+0	0.00E+0	0.00E+0	0.00E+0	0.00E+0	0.00E+0	0.00E+0	0.00E+0	0.00E+0	0.00E+0
PDS-19O	0.00E+0	0.00E+0	0.00E+0	0.00E+0	0.00E+0	0.00E+0	1.64E-7	1.83E-8	0.00E+0	0.00E+0	0.00E+0	0.00E+0	0.00E+0	0.00E+0	0.00E+0	0.00E+0	0.00E+0	0.00E+0	0.00E+0	0.00E+0	0.00E+0
Total	0.00E+0	1.32E-6	8.79E-7	1.05E-6	0.00E+0	0.00E+0	1.64E-7	1.83E-8	0.00E+0	0.00E+0	0.00E+0	0.00E+0	0.00E+0	0.00E+0	0.00E+0	0.00E+0	4.67E-8	5.19E-9	0.00E+0	0.00E+0	0.00E+0

Table 2-3
TMI-1 RADIONUCLIDE RELEASE CATEGORY FREQUENCY BY PLANT DAMAGE STATE (page 2 of 2)

	RC408	RC501	RC502	RC601	RC602	RC603	RC604	RC605	RC606	RC607	RC608	RC701	RC702	RC703	RC704	RC801	RC901	RC902	RC903	RC904	Total
PDS-01A	0.00E+0	5.45E-9	2.87E-10	3.71E-11	3.24E-14	1.93E-12	0.00E+0	1.94E-12	0.00E+0	9.91E-14	0.00E+0	3.52E-10	3.41E-13	1.85E-11	1.62E-14	4.37E-9	8.29E-8	8.29E-11	0.00E+0	0.00E+0	9.35E-8
PDS-01C	0.00E+0	0.00E+0	1.95E-11	0.00E+0	0.00E+0	0.00E+0	0.00E+0	4.52E-13	1.05E-10	2.34E-14	5.49E-12	0.00E+0	0.00E+0	4.28E-12	9.99E-10	7.47E-11	0.00E+0	0.00E+0	3.53E-12	3.54E-14	1.21E-9
PDS-02B	0.00E+0	0.00E+0	2.48E-9	0.00E+0	0.00E+0	0.00E+0	0.00E+0	0.00E+0	5.60E-9	0.00E+0	2.91E-10	0.00E+0	0.00E+0	0.00E+0	5.30E-8	7.19E-9	0.00E+0	0.00E+0	0.00E+0	0.00E+0	6.85E-8
PDS-03A	0.00E+0	1.42E-10	7.46E-12	9.65E-13	8.43E-16	5.01E-14	0.00E+0	5.05E-14	0.00E+0	2.58E-15	0.00E+0	9.14E-12	8.87E-15	4.81E-13	4.20E-16	1.13E-10	2.16E-9	2.16E-12	0.00E+0	0.00E+0	2.43E-9
PDS-04A	0.00E+0	5.95E-10	3.13E-11	4.05E-12	3.18E-15	2.08E-13	0.00E+0	2.12E-13	0.00E+0	1.07E-14	0.00E+0	3.84E-11	3.69E-14	2.02E-12	1.59E-15	4.76E-10	9.05E-9	9.05E-12	0.00E+0	0.00E+0	1.02E-8
PDS-04B	0.00E+0	0.00E+0	6.47E-10	0.00E+0	0.00E+0	0.00E+0	0.00E+0	0.00E+0	8.38E-10	0.00E+0	4.36E-11	0.00E+0	0.00E+0	0.00E+0	7.94E-9	1.08E-9	0.00E+0	0.00E+0	0.00E+0	0.00E+0	1.05E-8
PDS-04C	0.00E+0	0.00E+0	3.24E-11	0.00E+0	0.00E+0	0.00E+0	0.00E+0	7.51E-13	1.75E-10	3.76E-14	9.12E-12	0.00E+0	0.00E+0	7.11E-12	1.66E-9	1.24E-10	0.00E+0	0.00E+0	5.86E-12	5.84E-14	2.01E-9
PDS-04F	0.00E+0	0.00E+0	2.04E-11	0.00E+0	0.00E+0	0.00E+0	0.00E+0	3.84E-11	3.87E-13	2.00E-12	2.02E-14	0.00E+0	0.00E+0	3.63E-10	3.67E-12	7.12E-10	0.00E+0	0.00E+0	7.06E-10	7.12E-12	1.85E-9
PDS-04L	0.00E+0	0.00E+0	0.00E+0	0.00E+0	0.00E+0	0.00E+0	0.00E+0	0.00E+0	0.00E+0	0.00E+0	0.00E+0	0.00E+0	0.00E+0	0.00E+0	0.00E+0	0.00E+0	0.00E+0	0.00E+0	0.00E+0	0.00E+0	0.00E+0
PDS-05A	0.00E+0	9.44E-9	4.97E-10	6.43E-11	5.62E-14	3.32E-12	0.00E+0	3.37E-12	0.00E+0	1.72E-13	0.00E+0	6.09E-10	5.91E-13	3.21E-11	2.80E-14	7.57E-9	1.44E-7	1.44E-10	0.00E+0	0.00E+0	1.62E-7
PDS-05B	0.00E+0	0.00E+0	4.25E-10	0.00E+0	0.00E+0	0.00E+0	0.00E+0	0.00E+0	9.59E-10	0.00E+0	4.99E-11	0.00E+0	0.00E+0	0.00E+0	9.08E-9	1.23E-9	0.00E+0	0.00E+0	0.00E+0	0.00E+0	1.17E-8
PDS-05C	0.00E+0	0.00E+0	2.10E-10	0.00E+0	0.00E+0	0.00E+0	0.00E+0	2.90E-11	1.24E-9	1.51E-12	6.44E-11	0.00E+0	0.00E+0	2.76E-10	1.17E-8	1.45E-9	0.00E+0	0.00E+0	6.08E-10	6.14E-12	1.56E-8
PDS-05F	0.00E+0	0.00E+0	1.05E-8	0.00E+0	0.00E+0	0.00E+0	0.00E+0	8.85E-8	8.94E-10	4.61E-9	4.64E-11	0.00E+0	0.00E+0	8.38E-7	8.46E-9	0.00E+0	0.00E+0	0.00E+0	0.00E+0	0.00E+0	9.51E-7
PDS-06A	0.00E+0	8.28E-10	4.36E-11	5.64E-12	4.93E-15	2.91E-13	0.00E+0	2.95E-13	0.00E+0	1.51E-14	0.00E+0	5.34E-11	5.18E-14	2.81E-12	2.46E-15	6.63E-10	1.26E-8	1.26E-11	0.00E+0	0.00E+0	1.42E-8
PDS-07A	0.00E+0	3.17E-7	1.67E-8	2.25E-9	1.95E-12	1.08E-11	0.00E+0	1.18E-10	0.00E+0	5.11E-13	0.00E+0	2.04E-8	1.97E-11	1.08E-9	9.29E-13	2.54E-7	4.82E-6	4.82E-9	0.00E+0	0.00E+0	5.43E-6
PDS-07C	0.00E+0	0.00E+0	4.94E-8	0.00E+0	0.00E+0	0.00E+0	0.00E+0	1.20E-8	2.54E-7	5.93E-11	1.26E-9	0.00E+0	0.00E+0	1.08E-7	2.30E-6	2.54E-7	0.00E+0	0.00E+0	8.93E-8	9.00E-10	3.07E-6
PDS-07D	0.00E+0	2.46E-9	1.30E-10	1.14E-8	1.13E-11	5.67E-11	4.38E-14	6.02E-10	5.72E-13	2.95E-12	0.00E+0	1.03E-7	1.03E-10	5.44E-9	5.38E-12	0.00E+0	0.00E+0	0.00E+0	0.00E+0	0.00E+0	1.24E-7
PDS-07E	0.00E+0	0.00E+0	6.87E-12	0.00E+0	0.00E+0	0.00E+0	0.00E+0	0.00E+0	2.84E-11	0.00E+0	1.41E-13	0.00E+0	0.00E+0	0.00E+0	2.57E-10	3.50E-11	0.00E+0	0.00E+0	0.00E+0	0.00E+0	3.27E-10
PDS-07F	0.00E+0	0.00E+0	2.89E-8	0.00E+0	0.00E+0	0.00E+0	0.00E+0	2.36E-8	2.09E-7	1.17E-10	1.04E-9	0.00E+0	0.00E+0	2.14E-7	1.89E-6	2.52E-7	0.00E+0	0.00E+0	1.00E-8	1.01E-10	2.63E-6
PDS-07L	0.00E+0	0.00E+0	5.77E-10	0.00E+0	0.00E+0	0.00E+0	0.00E+0	0.00E+0	0.00E+0	0.00E+0	0.00E+0	0.00E+0	0.00E+0	0.00E+0	0.00E+0	0.00E+0	0.00E+0	0.00E+0	0.00E+0	0.00E+0	5.25E-8
PDS-08A	0.00E+0	4.78E-7	2.52E-8	3.40E-9	2.95E-12	1.63E-11	0.00E+0	1.79E-10	0.00E+0	7.72E-13	0.00E+0	3.08E-8	2.97E-11	1.62E-9	1.40E-12	3.83E-7	7.28E-6	7.28E-9	0.00E+0	0.00E+0	8.20E-6
PDS-08B	0.00E+0	0.00E+0	1.05E-7	0.00E+0	0.00E+0	0.00E+0	0.00E+0	0.00E+0	2.47E-7	0.00E+0	1.23E-9	0.00E+0	0.00E+0	0.00E+0	2.24E-6	3.04E-7	0.00E+0	0.00E+0	0.00E+0	0.00E+0	2.89E-6
PDS-08C	0.00E+0	0.00E+0	6.01E-8	0.00E+0	0.00E+0	0.00E+0	0.00E+0	8.68E-9	3.69E-7	4.26E-11	1.83E-9	0.00E+0	0.00E+0	7.88E-8	3.34E-6	4.14E-7	0.00E+0	0.00E+0	1.74E-7	1.75E-9	4.45E-6
PDS-08D	0.00E+0	1.68E-9	8.87E-11	7.81E-9	7.75E-12	3.88E-11	3.33E-14	4.11E-10	3.91E-13	2.03E-12	0.00E+0	7.06E-8	7.06E-11	3.72E-9	3.69E-12	0.00E+0	0.00E+0	0.00E+0	0.00E+0	0.00E+0	8.44E-8
PDS-08E	0.00E+0	0.00E+0	6.30E-10	0.00E+0	0.00E+0	0.00E+0	0.00E+0	0.00E+0	2.60E-9	0.00E+0	1.29E-11	0.00E+0	0.00E+0	0.00E+0	2.36E-8	3.21E-9	0.00E+0	0.00E+0	0.00E+0	0.00E+0	3.00E-8
PDS-10A	0.00E+0	1.07E-8	5.62E-10	7.26E-11	0.00E+0	3.67E-12	0.00E+0	3.70E-12	0.00E+0	1.53E-13	0.00E+0	6.88E-10	5.28E-13	3.61E-11	0.00E+0	8.55E-9	1.62E-7	1.62E-10	0.00E+0	0.00E+0	1.45E-6
PDS-10C	0.00E+0	0.00E+0	1.78E-10	0.00E+0	0.00E+0	0.00E+0	0.00E+0	0.00E+0	1.16E-9	0.00E+0	6.04E-11	0.00E+0	0.00E+0	0.00E+0	1.10E-8	7.84E-10	0.00E+0	0.00E+0	0.00E+0	0.00E+0	1.04E-7
PDS-12C	0.00E+0	0.00E+0	4.82E-11	0.00E+0	0.00E+0	0.00E+0	0.00E+0	0.00E+0	2.63E-10	0.00E+0	1.37E-11	0.00E+0	0.00E+0	0.00E+0	2.49E-9	1.78E-10	0.00E+0	0.00E+0	0.00E+0	0.00E+0	1.40E-8
PDS-15A	0.00E+0	0.00E+0	0.00E+0	0.00E+0	0.00E+0	0.00E+0	0.00E+0	0.00E+0	0.00E+0	0.00E+0	0.00E+0	0.00E+0	0.00E+0	0.00E+0	0.00E+0	0.00E+0	0.00E+0	0.00E+0	0.00E+0	0.00E+0	1.30E-7
PDS-18B	0.00E+0	0.00E+0	0.00E+0	0.00E+0	0.00E+0	0.00E+0	0.00E+0	0.00E+0	0.00E+0	0.00E+0	0.00E+0	0.00E+0	0.00E+0	0.00E+0	0.00E+0	0.00E+0	0.00E+0	0.00E+0	0.00E+0	0.00E+0	0.00E+0
PDS-18C	0.00E+0	0.00E+0	0.00E+0	0.00E+0	0.00E+0	0.00E+0	0.00E+0	0.00E+0	0.00E+0	0.00E+0	0.00E+0	0.00E+0	0.00E+0	0.00E+0	0.00E+0	0.00E+0	0.00E+0	0.00E+0	0.00E+0	0.00E+0	1.74E-6
PDS-18E	0.00E+0	0.00E+0	0.00E+0	0.00E+0	0.00E+0	0.00E+0	0.00E+0	0.00E+0	0.00E+0	0.00E+0	0.00E+0	0.00E+0	0.00E+0	0.00E+0	0.00E+0	0.00E+0	0.00E+0	0.00E+0	0.00E+0	0.00E+0	0.00E+0
PDS-18F	0.00E+0	0.00E+0	0.00E+0	0.00E+0	0.00E+0	0.00E+0	0.00E+0	0.00E+0	0.00E+0	0.00E+0	0.00E+0	0.00E+0	0.00E+0	0.00E+0	0.00E+0	0.00E+0	0.00E+0	0.00E+0	0.00E+0	0.00E+0	2.22E-8
PDS-19O	0.00E+0	0.00E+0	0.00E+0	0.00E+0	0.00E+0	0.00E+0	0.00E+0	0.00E+0	0.00E+0	0.00E+0	0.00E+0	0.00E+0	0.00E+0	0.00E+0	0.00E+0	0.00E+0	0.00E+0	0.00E+0	0.00E+0	0.00E+0	1.83E-7
Total	0.00E+0	8.26E-7	3.02E-7	2.51E-8	2.41E-11	1.32E-10	7.70E-14	1.34E-7	1.09E-6	4.84E-9	5.96E-9	2.27E-7	2.25E-10	1.25E-6	9.90E-6	1.90E-6	1.25E-5	1.25E-8	2.74E-7	2.77E-9	3.19E-5

Notes to Table 2-3:

Forty-one (41) release categories are used in the TMI Level 2 PRA. A numbering scheme is used to separate major categories:

1: Containment Bypass with Auxiliary Building Bypass
2: Interfacing-Systems LOCA
3: Large Isolation Failures
4: Small Isolation Failures
5: Early Containment Failure
6: Late Containment Failure (Catastrophic)
7: Late Containment Failure (Benign)
8: Basemat Melt-Through
9: No Containment Failure

The general characteristics of the individual release categories are described below:

1. Release Category 1.01: containment bypass, outside the auxiliary building, without ex-vessel release of fission products, with fission product scrubbing, release begins at approximately 4 hrs

2. Release Category 1.02: containment bypass, outside the auxiliary building, without ex-vessel release of fission products, without fission product scrubbing, release begins at approximately 3 hrs

3. Release Category 1.03: containment bypass, outside the auxiliary building, with ex-vessel release of fission products, with fission product scrubbing, release begins at approximately 4 hrs

4. Release Category 1.04: containment bypass, outside the auxiliary building, with ex-vessel release of fission products, without fission product scrubbing, release begins at approximately 3 hrs

5. Release Category 2.01: containment bypass, to the auxiliary building, without ex-vessel release of fission products, with fission product scrubbing, release begins at approximately 4 hrs

6. Release Category 2.02: containment bypass, to the auxiliary building, without ex-vessel release of fission products, without fission product scrubbing, release begins at approximately 3 hrs

7. Release Category 2.03: containment bypass, to the auxiliary building, with ex-vessel release of fission products, with fission product scrubbing, release begins at approximately 4 hrs

8. Release Category 2.04: containment bypass, to the auxiliary building, with ex-vessel release of fission products, without fission product scrubbing, release begins at approximately 3 hrs

9. Release Category 3.01: large isolation failure, to the auxiliary building, without ex-vessel release of fission products, with fission product scrubbing, release begins at approximately 1.5 hrs

10. Release Category 3.02: large isolation failure, to the auxiliary building, without ex-vessel release of fission products, without fission product scrubbing, release begins at approximately 1.5 hrs

11. Release Category 3.03: large isolation failure, to the auxiliary building, with ex-vessel release of fission products, with fission product scrubbing, release begins at approximately 1.5 hrs

12. Release Category 3.04: large isolation failure, to the auxiliary building, with ex-vessel release of fission products, without fission product scrubbing, release begins at approximately 1.5 hrs

13. Release Category 3.05: large isolation failure, outside the auxiliary building, without ex-vessel release of fission products, release begins at approximately 1.5 hrs

14. Release Category 3.06: large isolation failure, outside the auxiliary building, with ex-vessel release of fission products, release begins at approximately 1.5 hrs

15. Release Category 4.01: small isolation failure, to the auxiliary building, without ex-vessel release of fission products, with fission product scrubbing, release begins at approximately 2.5 hrs

16. Release Category 4.02: small isolation failure, to the auxiliary building, without ex-vessel release of fission products, without fission product scrubbing, release begins at approximately 2.5 hrs

17. Release Category 4.03: small isolation failure, to the auxiliary building, with ex-vessel release of fission products, with fission product scrubbing, release begins at approximately 2.5 hrs

18. Release Category 4.04: small isolation failure, to the auxiliary building, with ex-vessel release of fission products, without fission product scrubbing, release begins at approximately 2.5 hrs

19. Release Category 4.05: small isolation failure, to the environment, without ex-vessel release of fission products, with fission product scrubbing, release begins at approximately 2.5 hrs

20. Release Category 4.06: small isolation failure, to the environment, without ex-vessel release of fission products, without fission product scrubbing, release begins at approximately 2.5 hrs

21. Release Category 4.07: small isolation failure, to the environment, with ex-vessel release of fission products, without fission product scrubbing, release begins at approximately 2.5 hrs

22. Release Category 4.08: small isolation failure, to the environment, with ex-vessel release of fission products, without fission product scrubbing, release begins at approximately 2.5 hrs

23. Release Category 5.01: early containment failure, without ex-vessel fission product release, release begins at approximately 3.25 hrs

24. Release Category 5.02: early containment failure, with ex-vessel fission product release, release begins at approximately 5.5 hrs

25. Release Category 6.01: late overpressurization, with catastrophic containment failure, without ex-vessel fission product release, without revaporization, with fission product scrubbing, release begins at approximately 45 hrs

26. Release Category 6.02: late overpressurization, with catastrophic containment failure, without ex-vessel fission product release, without revaporization, without fission product scrubbing, release begins at approximately 45 hrs

27. Release Category 6.03: late overpressurization, with catastrophic containment failure, without ex-vessel fission product release, with revaporization, with fission product scrubbing, release begins at approximately 45 hrs

28. Release Category 6.04: late overpressurization, with catastrophic containment failure, without ex-vessel fission product release, with revaporization, without fission product scrubbing, release begins at approximately 45 hrs

29. Release Category 6.05: late overpressurization, with catastrophic containment failure, with ex-vessel release of fission products, without revaporization, with fission product scrubbing, release begins at approximately 45 hrs

30. Release Category 6.06: late overpressurization, with catastrophic containment failure, with ex-vessel release of fission products, without revaporization, without fission product scrubbing, release begins at approximately 45 hrs

31. Release Category 6.07: late overpressurization, with catastrophic containment failure, with ex-vessel release of fission products, with revaporization, with fission product scrubbing, release begins at approximately 45 hrs

32. Release Category 6.08: late overpressurization, with catastrophic containment failure, with ex-vessel release of fission products, with revaporization, without fission product scrubbing, release begins at approximately 45 hrs

33. Release Category 7.01: late overpressurization, with benign containment failure, without ex-vessel fission product release, with fission product scrubbing, release begins at approximately 14.5 hrs

34. Release Category 7.02: late overpressurization, with benign containment failure, without ex-vessel fission product release, without fission product scrubbing, release begins at approximately 14.5 hrs

35. Release Category 7.03: late overpressurization, with benign containment failure, with ex-vessel release of fission products, with fission product scrubbing, release begins at approximately 14.5 hrs

36. Release Category 7.04: late overpressurization, with benign containment failure, with ex-vessel release of fission products, without fission product scrubbing, release begins at approximately 14.5 hrs

37. Release Category 8.01: containment failure from basemat melt-through, with ex-vessel release of fission products, release begins at approximately 36 hrs

38. Release Category 9.01: no containment failure, without ex-vessel fission product release, with fission product scrubbing, release begins at approximately 0.5 hrs

39. Release Category 9.02: no containment failure, without ex-vessel fission product release, without fission product scrubbing, release begins at approximately 2.5 hrs

40. Release Category 9.03: no containment failure, with ex-vessel fission product release, with fission product scrubbing, release begins at approximately 2.5 hrs

41. Release Category 9.04: no containment failure, with ex-vessel fission product release, without fission product scrubbing, release begins at approximately 2.5 hrs

2.4.4 TMI-1 IPEEE

As discussed in Section 2.3, the TMI-1 internal events Level 1 model and the Level 2 results [25] are used as the primary basis for this risk assessment. However, external event risk results from the TMI-1 Individual Plant Examination of External Events (IPEEE) are used in this analysis in a sensitivity discussion to confirm that the conclusion of this analysis does not change if external events are included in the decision making process.

Refer to Appendix A of this report for further details of the TMI-1 IPEEE and the use of the IPEEE results in this risk assessment.

Table 2-4

TMI-1 50-MILE RADIUS POPULATION DOSE AND DOSE RATE AS A FUNCTION OF TMI-1 PRA RELEASE CATEGORY [22]

Release Category	Release Category Frequency (1/yr)	50-Mile Radius Population Dose (Person-Rem)	50-Mile Radius Population Dose Rate (Person-Rem/yr)
RC101	0.00E+00	5.05E+05	0.00E+00
RC102	1.32E-06	1.01E+06	1.34E+00
RC103	8.79E-07	5.06E+05	4.45E-01
RC104	1.05E-06	1.01E+06	1.06E+00
RC201	0.00E+00	3.53E+06	0.00E+00
RC202	0.00E+00	1.18E+07	0.00E+00
RC203	1.64E-07	3.71E+06	6.10E-01
RC204	1.83E-08	1.34E+07	2.45E-01
RC301	0.00E+00	1.56E+06	0.00E+00
RC302	0.00E+00	3.21E+06	0.00E+00
RC303	0.00E+00	1.89E+06	0.00E+00
RC304	0.00E+00	4.05E+06	0.00E+00
RC305	0.00E+00	3.21E+06	0.00E+00
RC306	0.00E+00	4.03E+06	0.00E+00
RC401	0.00E+00	4.23E+05	0.00E+00
RC402	0.00E+00	1.18E+06	0.00E+00
RC403	4.67E-08	6.10E+05	2.85E-02
RC404	5.19E-09	1.57E+06	8.15E-03
RC405	0.00E+00	5.42E+05	0.00E+00
RC406	0.00E+00	1.41E+06	0.00E+00
RC407	0.00E+00	9.47E+05	0.00E+00
RC408	0.00E+00	2.13E+06	0.00E+00
RC501	8.26E-07	1.15E+06	9.50E-01
RC502	3.02E-07	1.21E+06	3.66E-01
RC601	2.51E-08	3.30E+05	8.28E-03
RC602	2.41E-11	8.23E+05	1.98E-05
RC603	1.32E-10	1.15E+06	1.52E-04
RC604	7.70E-14	1.68E+06	1.29E-07
RC605	1.34E-07	3.37E+05	4.52E-02
RC606	1.09E-06	8.49E+05	9.28E-01
RC607	4.84E-09	1.16E+06	5.62E-03

Table 2-4

TMI-1 50-MILE RADIUS POPULATION DOSE AND DOSE RATE AS A FUNCTION OF TMI-1 PRA RELEASE CATEGORY [22]

Release Category	Release Category Frequency (1/yr)	50-Mile Radius Population Dose (Person-Rem)	50-Mile Radius Population Dose Rate (Person-Rem/yr)
RC608	5.96E-09	1.69E+06	1.01E-02
RC701	2.27E-07	1.27E+05	2.88E-02
RC702	2.25E-10	4.21E+05	9.47E-05
RC703	1.25E-06	1.28E+05	1.60E-01
RC704	9.90E-06	4.23E+05	4.19E+00
RC801	1.90E-06	5.78E+04	1.10E-01
RC901	1.25E-05	3.53E+02	4.41E-03
RC902	1.25E-08	8.67E+03	1.08E-04
RC903	2.74E-07	2.00E+03	5.48E-04
RC904	2.77E-09	8.76E+03	2.42E-05

2.4.5 TMI-1 Past ILRT Results

The surveillance frequency for Type A testing in NEI 94-01 under option B criteria is at least once per ten years based on an acceptable performance history (i.e., two consecutive periodic Type A tests at least 24 months apart where the calculated performance leakage rate was less than 1.0 L_a) and consideration of the performance factors in NEI 94-01, Section 11.3.

In June 1996 revised TMI-1 Technical Specifications implementing the performance-based leakage rate testing program were submitted to the NRC for approval. In May 1997, the NRC issued Amendment 201 to the TMI-1 Technical Specifications. Based on completion of two successful ILRTs at TMI-1, the current ILRT interval is once per ten years. The next Type A test for TMI-1 would have been required to be completed by October 2003 [16]. However, a subsequent licensing submittal that extended the ILRT to a one-time 15-year interval was approved [24], thus requiring performance of the Type A ILRT by October 2008, which would require a mid-cycle outage in order to perform.

Note that the probability of a pre-existing leakage due to extending the ILRT interval is based on the industry wide historical results as discussed in the NEI Guidance document, and the only specific TMI information utilized is the fact that the current one-time ILRT interval is once per 15 years.

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 TMI-1 Level 1 and Level 2 PRA results. The results for each EPRI category are described below.

To calculate the various release category frequencies and aid in the documenting and tabulation of results, an Excel spreadsheet was employed to perform the various steps described in the section [34].

<u>Frequency of EPRI Category 1</u>

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 RC9 in the TMI-1 Level 2 PRA).

As discussed previously in Section 2.4.2, the frequency of TMI-1 severe accidents in which the containment remains intact is 1.28E-5/yr. As described below, the frequencies of the 3a and 3b categories are (3.89E-7/yr + 3.49E-7/yr) and (3.89E-8/yr + 3.49E-8/yr), respectively. Therefore, the frequency of EPRI Category 1 is calculated as (1.28E-5/yr) – (7.38E-7/yr + 7.38E-8/yr) = 1.20E-5/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 determined by summing the frequencies of the RC3-series and RC4-series release categories. As discussed in the notes to Table 2-3, the RC3-series and the RC4-series release categories represent severe accidents with containment isolation failure.

From Table 2-3, the frequency of EPRI Category 2 is (12 x 0.00) + 4.67E-8/yr + 5.19E-9/yr = 5.19E-8/yr.

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 that would be identifiable only from an ILRT (and thus affected by ILRT testing frequency).

The base NEI Interim Guidance describes the calculation of a screening frequency for Category 3a and 3b by multiplying the entire plant CDF by a pre-existing containment leakage probability. In supplemental guidance provided in Reference [21], NEI discusses the conservative nature of the screening frequency calculation and describes which CDF sequence contributors can be removed from the total plant CDF to obtain a less conservative frequency estimate. Consistent with the supplement NEI Interim Guidance provided in Reference [21], the frequency per year for this category is calculated as:

Frequency 3a = [3a conditional failure probability] x [CDF - (CDF with containment failure independent of containment leakage)]

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 involve containment failure independent of potential pre-existing containment leakage. The following core damage accidents involve containment failure or bypass regardless of the potential existence of pre-existing containment leakage:

- Containment Bypass accidents (TMI-1 PRA RC1 and RC2)
- Severe accidents with Containment Isolation System failure (TMI-1 PRA RC3 and RC4)
- Severe accidents with containment failure due to energetic phenomena (TMI-1 PRA RC5)

Therefore, the TMI-1 PRA RC6-series, RC7-series, RC8-series and RC9-series release categories are used as the CDF on which to apply the 3a conditional failure probability.

In addition to the above, consistent with the supplemental NEI guidance in Reference [21], EPRI Category 3a is refined in this risk assessment into accidents with containment sprays available (3a Scrubbed) and with sprays unavailable (3a Unscrubbed). This refinement requires separating the RC6, RC7, RC8, and RC9 accidents into scrubbed and unscrubbed. Using the TMI-1 Level 2 PRA information summarized in Table 2-3, the contribution of scrubbed and unscrubbed sequences to RC6-9 release categories is as follows:

Release Category	Frequency of Scrubbed Accident Sequences (1/yr)	Frequency of Unscrubbed Accident Sequences (1/yr)
RC6-series	1.64E-07	1.10E-06
RC7-series	1.48E-06	9.90E-06
RC8-series	0.00E+00	1.90E-06
RC9-series	1.28E-05	1.53E-08
TOTAL:	1.44E-05	1.29E-05

Therefore, the frequency of category 3a (Scrubbed) is calculated as (2.70E-02) x (1.44E-05/yr) = 3.89E-07/yr. Likewise, the frequency of category 3a (Unscrubbed) is calculated as (2.70E-02) x (1.29E-05/yr) = 3.49E-07/yr.

<u>Frequency of EPRI Category 3b</u>

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 that would be identifiable only from an ILRT (and thus affected by ILRT testing frequency). In the same manner as that discussed previously for category 3a, EPRI Category 3b is refined into accidents with containment sprays available (3b Scrubbed) and with sprays unavailable (3b Unscrubbed). The frequencies are calculated in the same manner as that described above for 3a, except that the 3b pre-existing leakage probability is 2.7E-3. This 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, the frequency of category 3b (Scrubbed) is calculated as (2.70E-03) x (1.44E-05/yr) = 3.89E-08/yr. Likewise, the frequency of category 3b (Unscrubbed) is calculated as (2.70E-03) x (1.29E-05/yr) = 3.49E-08/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 discussed in Section 2.4.2, the TMI-1 Level 2 PRA categorizes severe accidents into nine release categories (RC1 through RC9). TMI-1 containment bypass scenarios (RC1 and RC2) are assigned to EPRI Category 8. TMI-1 containment isolation system failure scenarios (RC3 and RC4) are assigned to EPRI Category 2. TMI-1 severe accidents with an intact containment (RC9) are assigned to EPRI Category 1. The remaining spectrum of TMI-1 severe accidents (RC5, RC6, RC7, and RC8) applies to EPRI Category 7.

Therefore, the frequency of EPRI Category 7 is calculated as 1.13E-6/yr + 1.26E-6/yr + 1.14E-5/yr + 1.90E-6/yr = 1.57E-5/yr.

<u>Frequency of EPRI Category 8</u>

This group consists of all core damage accident progression bins in which the accident is initiated by a containment bypass scenario (i.e., Steam Generator Tube Rupture or Interfacing Systems LOCA, ISLOCA). The frequency of Category 8 is determined by summing the frequencies of the RC1-series and RC2-series release categories. As discussed in the notes to Table 2-3, the RC1-series and RC2-series release categories represent severe accidents initiated by a containment bypass event. Therefore, the frequency of Category 8 is calculated as (3 x 0.0) + 1.32E-6 + 8.79E-7 + 1.05E-6 + 1.64E-7 + 1.83E-8 = 3.44E-6/yr.

<u>Summary of Frequencies of EPRI Accident Categories</u>

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.

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. The maximum allowable overall containment leak rate (at all times between required Type A testing) for TMI-1 is less than or equal to 0.1 weight percent of the containment atmosphere per 24 hours at accident pressure (P_{AC}).

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.

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.	Per NEI Interim Guidance: *[Sum of TMI-1 release category RC9 frequencies] – [Frequency EPRI Categories 3a and 3b]* *[1.28E-5/yr] – [7.38E-7/yr + 7.38E-8/yr] = 1.20E-5/yr*	1.20E-05
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.	Per NEI Interim Guidance: *[Sum of TMI-1 release category RC3 and RC4 frequencies]*	5.19E-08
3a (Scrubbed)	Small Pre-Existing Failures (Scrubbed): Accident sequences in which the containment is failed due to a pre-existing small leak in the containment structure that would be identifiable only from an ILRT (and thus affected by ILRT testing frequency). Releases are scrubbed.	Per NEI Interim Guidance: *[Sum of TMI-1 frequencies for RC6 thru RC9 "scrubbed" accidents] x [2.7E-2]* *[1.44E-05/yr] x [2.70E-02] = 3.89E-7/yr*	3.89E-07
3a (Unscrubbed)	Small Pre-Existing Failures (Unscrubbed): Accident sequences in which the containment is failed due to a pre-existing small leak in the containment structure that would be identifiable only from an ILRT (and thus affected by ILRT testing frequency). Releases are unscrubbed.	Per NEI Interim Guidance: *[Sum of TMI-1 frequencies for RC6 thru RC9 "unscrubbed" accidents] x [2.7E-2]* *[1.29E-05/yr] x [2.70E-02] = 3.49E-7/yr*	3.49E-07

Table 3-1

BASELINE RELEASE FREQUENCY AS A FUNCTION OF EPRI CATEGORY

EPRI Category	Category Description	Frequency Estimation Methodology	Frequency (1/yr)
3b (Scrubbed)	Large Pre-Existing Failures (Scrubbed): Accident sequences in which the containment is failed due to a pre-existing large leak in the containment structure that would be identifiable only from an ILRT (and thus affected by ILRT testing frequency). Releases are scrubbed.	Per NEI Interim Guidance: *[Sum of TMI-1 frequencies for RC6 thru RC9 "scrubbed" accidents] x [2.7E-3]* *[1.44E-05/yr] x [2.70E-03] = 3.89E-8/yr*	3.89E-08
3b (Unscrubbed)	Large Pre-Existing Failures (Unscrubbed): Accident sequences in which the containment is failed due to a pre-existing large leak in the containment structure that would be identifiable only from an ILRT (and thus affected by ILRT testing frequency). Releases are unscrubbed.	Per NEI Interim Guidance: *[Sum of TMI-1 frequencies for RC6 thru RC9 "unscrubbed" accidents] x [2.7E-3]* *[1.29E-05/yr] x [2.70E-03] = 3.49E-8/yr*	3.49E-08
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

Table 3-1

BASELINE RELEASE FREQUENCY AS A FUNCTION OF EPRI CATEGORY

EPRI Category	Category Description	Frequency Estimation Methodology	Frequency (1/yr)
7	<u>Containment Failure Due to Severe Accident Progression</u>: EPRI Category 7 applies to accident sequences in which the containment is failed due to the severe accident progression. Not affected by ILRT leak testing frequency.	Per NEI Interim Guidance: *[Sum of TMI-1 release category RC5, RC6, RC7, and RC8 frequencies]*	1.57E-05
8	<u>Containment Bypass Accidents</u>: Accident sequences in which the containment is bypassed. Such accidents are initiated by LOCAs outside containment (i.e., SGTR or Interfacing Systems LOCA). Not affected by ILRT leak testing frequency.	Per NEI Interim Guidance: *[Sum of TMI-1 release category RC1 and RC2 frequencies]*	3.44E-06
		TOTAL:	3.19E-05

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)

As discussed in Section 2.4.3, population dose estimates used in this risk assessment are TMI-1 specific estimates taken from TMI Calculation No. C-1101-900-E-220-178.[22] The TMI-1 specific 50-mile radius doses by TMI-1 release category are summarized in Table 2-4. Use of the 50-mile radius population dose (i.e., as opposed to dose at the site boundary or some other radial distance) is consistent with previously approved ILRT submittals.

Using the TMI-1 dose information summarized in Table 2-4, the population dose by EPRI accident category is defined. Three general approaches to assigning representative doses to the EPRI categories may be followed:

1. Select a suitably representative accident sequence type (i.e., TMI-1 release sub-category) for each EPRI category.
2. Select the worst-core TMI-1 release sub-category for each EPRI category.
3. Use a weighted average (on a frequency contribution basis) of the constituent release categories for each EPRI category.

The third approach is used in this risk assessment after factoring in the unique results for all of the available release categories for TMI-1. The release category assignments are consistent with the approach utilized by Crystal River (that has a similar release categorization) in their ILRT extension request [20]. This allows for the most accuracy and refinement in performing the subsequent calculations. The weighted average dose is then calculated as: (sum of constituent RC dose rates) / (sum of constituent RC frequencies). Refer to Table 2-4 for details of the TMI-1 PRA release category doses, dose rates and frequencies. The resulting baseline dose estimates as a function of EPRI category are summarized in Table 3-2.

The dose for the "no containment failure" EPRI category (EPRI Category 1) is based on the weighted average of the doses associated with the TMI-1 PRA RC9-series release categories. The RC9 series release categories represent severe accidents in which the containment is isolated and remains intact (i.e., no containment failure but containment leakage at the Technical Specification allowable leakage rate).

The dose for EPRI Category 2 (containment isolation failure) is based on the weighted average of the doses associated with the TMI-1 PRA RC3-series and RC4-series release categories. The RC3-series release categories represent severe accidents with a "small" area containment isolation failure. Similarly, the RC4-series release categories represent severe accidents with a "large" containment isolation failure.

As discussed in Section 3.1, consistent with guidance in Reference [21], EPRI Category 3a is refined into accidents with containment sprays available (3a Scrubbed) and with sprays unavailable (3a Unscrubbed). The base dose rate (i.e., containment leakage within Technical Specifications) is based on the doses associated with the TMI-1 PRA RC9-series release categories. The weighted average dose of the RC901 and RC903 release categories (which represent containment intact accidents with sprays available) is used for category 3a (Scrubbed). The weighted average dose of the RC902 and RC904 release categories (which represent containment intact accidents with sprays unavailable) is used for category 3a (Unscrubbed). Per the NEI Interim Guidance, the base dose rates for 3a (Scrubbed) and 3a (Unscrubbed) are multiplied by a factor of 10 to reflect the potential for a "small" pre-existing containment leakage pathway.

Table 3-2

TMI-1 BASELINE DOSE ESTIMATES AS A FUNCTION OF EPRI CATEGORY FOR POPULATION WITHIN 50-MILE RADIUS

EPRI Category	TMI-1 PRA Release Categories Used in Characterizing Dose			50-Mile Radius Dose Applied to EPRI Category (Person-Rem)(1)
	Release Category (RC)	Sum of RC Frequencies (1/yr)	Sum of 50-Mile Radius Population Dose Rates (Person-Rem/yr)	
1	RC9 (all)	1.28E-05	5.09E-03	3.98E+02
2	RC3 (all)	5.19E-08	3.66E-02	7.06E+05
	RC4 (all)			
3a (Scrubbed)	RC901	1.28E-05	4.96E-03	3.88E+03 (Note 2)
	RC903			
3a (Unscrubbed)	RC902	1.53E-08	1.33E-04	8.69E+04 (Note 2)
	RC904			
3b (Scrubbed)	RC901	1.28E-05	4.96E-03	1.36E+04 (Note 3)
	RC903			
3b (Unscrubbed)	RC902	1.53E-08	1.33E-04	3.04E+05 (Note 3)
	RC904			
4	N/A	N/A	N/A	N/A
5	N/A	N/A	N/A	N/A
6	N/A	N/A	N/A	N/A
7	RC5 (all)	1.57E-05	6.80E+00	4.34E+05
	RC6 (all)			
	RC7 (all)			
	RC8 (all)			
8	RC1 (all)	3.44E-06	3.70E+00	1.08E+06
	RC2 (all)			

NOTES:

(1) A weighted average approach of the constituent TMI-1 PRA release categories applicable to each EPRI accident category is used to define the dose for each EPRI category. The weighted average dose is calculated as: (sum of constituent RC dose rates) / (sum of constituent RC frequencies). Refer to Table 2-4 for details of the TMI-1 PRA release category doses, dose rates, and frequencies.

(2) Consistent with guidance in Reference [21], EPRI Category 3a is refined into accidents with containment sprays available (3a Scrubbed) and with sprays unavailable (3a Unscrubbed). The base dose rate (i.e., containment leakage within Technical Specifications) is calculated in the manner described above for Note (1); however, the base dose rate is multiplied by a factor of 10, per the NEI Interim Guidance, to reflect the potential for a "small" pre-existing containment leakage pathway.

(3) Same comment as Note (2), except that the base dose rate is multiplied by a factor of 35, per the NEI Interim Guidance, to reflect the potential for a "large" pre-existing containment leakage pathway.

The discussion above for the 3a EPRI category applies to the 3b category, as well; except that the base dose rate is multiplied by a factor of 35 to reflect the potential for a "large" pre-existing containment leakage pathway.

As EPRI Categories 4, 5, and 6 are not affected by ILRT frequency and not analyzed as part of this risk assessment (per NEI Interim Guidance), no assignment of doses is made for these categories.

The dose for EPRI Category 7 (accidents with containment failure due to severe accident progression) is based on the weighted average of the doses associated with the TMI-1 PRA RC5-series, RC6-series, RC7-series and RC8-series release categories. These release categories represent severe accidents with containment failure occurring as a result of the severe accident progression (e.g., steam explosion, overpressurization, basemat melt-through).

The dose for the containment bypass category (EPRI Category 8) is based on the weighted average of the doses associated with the TMI-1 PRA RC1-series (SGTR accidents) and RC2-series (ISLOCA accidents) release categories.

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 3-2 by the frequencies summarized in Table 3-1. The resulting baseline population dose rates by EPRI category are summarized in Table 3-3. 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 dose rates shown in Table 3-3 are indicative of a 3-per-10 year ILRT surveillance frequency.

Table 3-3

BASELINE DOSE RATE ESTIMATES BY EPRI ACCIDENT CATEGORY FOR POPULATION WITHIN 50-MILE RADIUS

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	3.98E+02	1.20E-05 (Note 1)	4.77E-03
2	Containment Isolation System Failure	7.06E+05	5.19E-08	3.66E-02
3a (Scrubbed)	Small Pre-Existing Failures (with fission product scrubbing)	3.88E+03	3.89E-07	1.51E-03
3a (Unscrubbed)	Small Pre-Existing Failures (without fission product scrubbing)	8.69E+04	3.49E-07	3.03E-02
3b (Scrubbed)	Large Pre-Existing Failures (with fission product scrubbing)	1.36E+04	3.89E-08	5.29E-04
3b (Unscrubbed)	Large Pre-Existing Failures (without fission product scrubbing)	3.04E+05	3.49E-08	1.06E-02
4	Type B Failures	NA	NA	NA
5	Type C Failures	NA	NA	NA
6	Other Containment Isolation System Failures	NA	NA	NA
7	Containment Failure Due to Severe Accident Progression	4.34E+05	1.57E-05	6.80E+00
8	Containment Bypass Accidents	1.08E+06	3.44E-06	3.70E+00
		TOTAL:	3.19E-05	10.580

NOTES:

(1) The contribution of the EPRI category 3a and 3b frequencies are subtracted from the sum total of RC9 frequencies to avoid "double counting."

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-15 years, then the average time that a leak would be undetected is 90 months (surveillance interval of 180 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:

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

The base pre-existing ILRT-detectable leakage probabilities above are reflective of a 3-per-10 year ILRT frequency. The TMI-1 plant is currently operating under a one-time 1-per-15 year ILRT testing frequency [24]. The baseline 3-per-10 year based leakage probabilities first need to be adjusted to reflect the 1-per-15 year TMI-1 ILRT testing frequency as follows:

- "Small": 2.70E-2 x [(180 months/2) / (36 months/2)] = 1.35E-1
- "Large": 2.70E-3 x [(180 months/2) / (36 months/2)] = 1.35E-2

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-16.25 year ILRT frequency currently being pursued by TMI-1 (and the subject of this risk assessment) are calculated as follows:

- "Small": 1.35E-1 x [(195 months/2) / (180 months/2)] = 1.46E-1
- "Large": 1.35E-2 x [(195 months/2) / (180 months/2)] = 1.46E-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):

Table 3-4

EPRI CATEGORY FREQUENCY AS A FUNCTION OF ILRT INTERVAL (1/yr)

EPRI Category	Baseline (3-per-10 year ILRT)	Previous Extension (1-per-15 year ILRT)	Proposed Extension (1-per-16.25 year ILRT)
1	1.20E-05	8.74E-06	8.40E-06
3a (Scrubbed)	3.89E-07	1.95E-06	2.11E-06
3a (Unscrubbed)	3.49E-07	1.74E-06	1.89E-06
3b (Scrubbed)	3.89E-08	1.95E-07	2.11E-07
3b (Unscrubbed)	3.49E-08	1.74E-07	1.89E-07

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. Also, the sum of the 3a and 3b EPRI categories, which make use of release categories RC6 through RC9, above in Table 3-4 is subtracted from the overall frequency for the EPRI 1 category to avoid double counting. That is, the baseline frequency in Table 3-2 for EPRI category 1 is 1.28E-5. When subtracting the sum of the 3a and 3b categories in Table 3-4 (8.12E-7) from 1.28E-5, the result is 1.20E-5.

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 slightly from the current TMI-1 one-time 1-per-15 year ILRT interval to the proposed 1-per-16.25 year ILRT interval. The total dose increases from 10.750 person-rem/year to 10.768 person-rem/year (an increase of <0.2%).

Table 3-5

DOSE RATE ESTIMATES AS A FUNCTION OF ILRT INTERVAL
FOR POPULATION WITHIN 50-MILE RADIUS

EPRI Category	Category Description	Dose Rate as a Function of ILRT Interval (Person-Rem/Yr)		
		Baseline (3-per-10 year ILRT)	Current (1-per-15 year ILRT)	Proposed (1-per-16.25 year ILRT)
1	No Containment Failure	4.77E-03	3.48E-03	3.34E-03
2	Containment Isolation System Failure	3.66E-02	3.66E-02	3.66E-02
3a (Scrubbed)	Small Pre-Existing Failures (with fission product scrubbing)	1.51E-03	7.56E-03	8.19E-03
3a (Unscrubbed)	Small Pre-Existing Failures (without fission product scrubbing)	3.03E-02	1.51E-01	1.64E-01
3b (Scrubbed)	Large Pre-Existing Failures (with fission product scrubbing)	5.29E-04	2.65E-03	2.87E-03
3b (Unscrubbed)	Large Pre-Existing Failures (without fission product scrubbing)	1.06E-02	5.30E-02	5.74E-02
4	Type B Failures	NA	NA	NA
5	Type C Failures	NA	NA	NA
6	Other Containment Isolation System Failures	NA	NA	NA
7	Containment Failure Due to Severe Accident Progression	6.80E+00	6.80E+00	6.80E+00
8	Containment Bypass Accidents	3.70E+00	3.70E+00	3.70E+00
	TOTAL:	10.580	10.750	10.768

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 TMI-1 one-time 1-per-15 year ILRT interval, the percentage contribution to total dose rate from Categories 3a and 3b is shown to be very minor:

$$[(7.56E\text{-}3 + 1.51E\text{-}1 + 2.65E\text{-}3 + 5.30E\text{-}2) / 10.750] \times 100 = 2.0\%$$

For the proposed 1-per-16.25 year ILRT interval, the percentage contribution to total dose rate from Categories 3a and 3b increases slightly but remains minor:

$$[(8.19E\text{-}3 + 1.64E\text{-}1 + 2.87E\text{-}3 + 5.74E\text{-}2) / 10.768] \times 100 = 2.2\%$$

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. Category 3b is refined in this risk assessment into a scrubbed portion and an unscrubbed portion. The doses associated with 3b (Scrubbed) are not representative of large early releases as characterized by the LERF risk measure. As such, the change in LERF (Large Early Release Frequency) is determined by the change in the frequency of Category 3b (Unscrubbed).

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, 7, and 8 could result in large releases but these are not affected by the change in ILRT interval. Additionally, EPRI categories 4, 5, and 6 are not applicable.

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 (Unscrubbed) for 1-per-16.25 year ILRT interval)] –
[(Frequency of EPRI Category 3b (Unscrubbed) for 1-per-15 year ILRT interval)]

= 1.89E-7/yr – 1.74E-7/yr
= 1.45E-8/yr[1]

This delta LERF of 1.45E-8/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 TMI-1 from the currently allowed 1-per-15 years to 1-per-16.25 years represents a very small change in risk, and is an acceptable plant change from a risk perspective.

The cumulative impact on the LERF risk measure due to the proposed ILRT interval extension is calculated by comparing the 1-per-16.25 year ILRT interval with the baseline 3-per-10 year interval as follows:

delta LERF = [(Frequency of EPRI Category 3b (Unscrubbed) for 1-per-16.25 year ILRT interval)] –
[(Frequency of EPRI Category 3b (Unscrubbed) for 3-per-10 year ILRT interval)]

= 1.89E-7/yr – 3.49E-8/yr
= 1.54E-7/yr

[1] The 1.45E-8/yr value, as are all calculated values in this analysis, is determined using a spreadsheet calculation of summed frequencies that contain additional significant figures beyond the 2 digits shown in the two numbers subtracted above.

This delta LERF of 1.54E-7/yr falls into Region II, Small Change in Risk, of the acceptance guidelines in NRC Regulatory Guide 1.174. Even though this value is in excess of 1E-7, the total LERF was estimated to be 3.53E-06, which is in keeping with LERF being <1E-5 per reactor year (Region II) in accordance with the guidelines given in Section 2.2.4 of NRC Regulatory Guide 1.174. Therefore, increasing the ILRT interval at TMI-1 from the baseline 3-per-10 years to a one-time interval of 1-per-16.25 years represents a small cumulative 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 pre-existing leakages (EPRI Category 3a). The conditional part of the definition is conditional given a severe accident (i.e., core damage).

Consequently, the $CCFP_{\%}$ for a given ILRT interval can be calculated using the following equation:

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

For the 3-per-10 year interval:

$$CCFP_{Base} = [1 - ((1.20E\text{-}5 + 3.89E\text{-}7 + 3.49E\text{-}7) / 3.19E\text{-}5)] \times 100\%$$
$$= 60.2\%$$

For the one-time 15-year interval:

$$CCFP_{15} = [1 - ((8.74E\text{-}6 + 1.95E\text{-}6 + 1.74E\text{-}6) / 3.19E\text{-}5)] \times 100\%$$
$$= 61.1\%$$

And for a one-time 16.25-year interval:

$$CCFP_{16.25} = [1 - ((8.40E\text{-}6 + 2.11E\text{-}6 + 1.89E\text{-}6) / 3.19E\text{-}5)] \times 100\%$$
$$= 61.2\%$$

Therefore, the incremental change in the conditional containment failure probability is 0.1%, and the cumulative change is 1.0%. This cumulative change in $CCFP_{\%}$ of 1 percentage point is deemed small 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 16.25 years.

The analysis performed examined TMI-1 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

This analysis is performed using the TMI-1 internal events Level 1 and Level 2 PRAs. The quantitative results are summarized in Table 4-1. The key results to this risk assessment are those for the one-time 15-year interval (current TMI-1 condition) and the proposed one-time 16.25-year interval extension. The 3-per-10 year ILRT is a baseline starting point for this risk assessment given that the pre-existing containment leakage probabilities are reflective of the 3-per-10 year ILRT testing, estimated based on industry experience (refer to Section 3.1).

The following is a brief summary of some of the key aspects of the ILRT test interval extension risk analysis for a one-time increase from 15 to 16.25 years:

- Increasing the current one-time 15-year ILRT interval to 16.25 years results in an insignificant increase in total population dose rate, from 10.750 person-rem/year to 10.768 person-rem/year, respectively.
- The increase in the LERF risk measure, 1.45E-8/yr, is categorized as a "very small" increase per NRC Regulatory Guide 1.174.
- Likewise, the conditional containment failure probability ($CCFP_{\%}$) increases insignificantly by 0.1 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					
		Methodology Baseline (3-per-10 year ILRT)		Current Extension (1-per-15 year ILRT)		Proposed Extension (1-per-16.25 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	3.98E+02	1.20E-05	4.77E-03	8.74E-06	3.48E-03	8.40E-06	3.34E-03
2	7.06E+05	5.19E-08	3.66E-02	5.19E-08	3.66E-02	5.19E-08	3.66E-02
3a (Scrubbed)	3.88E+03	3.89E-07	1.51E-03	1.95E-06	7.56E-03	2.11E-06	8.19E-03
3a (Unscrubbed)	8.69E+04	3.49E-07	3.03E-02	1.74E-06	1.51E-01	1.89E-06	1.64E-01
3b (Scrubbed)	1.36E+04	3.89E-08	5.29E-04	1.95E-07	2.65E-03	2.11E-07	2.87E-03
3b (Unscrubbed)	3.04E+05	3.49E-08	1.06E-02	1.74E-07	5.30E-02	1.89E-07	5.74E-02
4	N/A	NA	N/A	NA	N/A	NA	N/A
5	N/A	NA	N/A	NA	N/A	NA	N/A
6	N/A	NA	N/A	NA	N/A	NA	N/A
7	4.38E+05	1.57E-05	6.80E+00	1.57E-05	6.80E+00	1.57E-05	6.80E+00
8	1.08E+06	3.44E-06	3.70E+00	3.44E-06	3.70E+00	3.44E-06	3.70E+00
TOTALS:		3.19E-05	10.580	3.19E-05	10.750	3.19E-05	10.768
Increase in Dose Rate (1)					1.7E-01		1.8E-02
Increase in LERF (2)				1.39E-07		1.45E-08	
Increase in $CCFP_{\%}$ (3)				0.9		0.1	

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-16.25 ILRT is calculated as: Total dose rate for 1-per-16.25 year ILRT, 10.768, minus total dose rate for 1-per-15 year ILRT, 10.750.

(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 (Unscrubbed). For example, the increase in LERF for the proposed 1-per-16.25 ILRT is calculated as: 3b (Unscrubbed) frequency for 1-per-16.25 year ILRT, 1.89E-07/yr, minus 3b (Unscrubbed) frequency for 1-per-15 year ILRT, 1.74E-07/yr.

(3) The increase in the conditional containment failure probability ($CCFP_{\%}$) is with respect to the results for the preceding ILRT interval, as presented in the table. As discussed in Section 3.4.5, $CCFP_{\%}$ is calculated as:

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

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 incremental impact on plant risk associated with extending the Type A ILRT test frequency from 15 years to 16.25 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-15 years to once-per-16.25 years is 1.45E-8/yr. Guidance in Reg. Guide 1.174 defines very small changes in LERF as below 10^{-7}/yr. Therefore, increasing the TMI-1 ILRT interval from 15 to 16.25 years results in a very small change in risk, and is an acceptable plant change from a risk perspective.

Based on the results from Sections 3 and 4, the main conclusion regarding the cumulative impact on plant risk associated with extending the Type A ILRT test frequency from 3-per-10 years to 16.25 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 small changes in risk as resulting in increases of CDF below 10^{-5}/yr and

increases in LERF below 10^{-6}/yr. Since the ILRT does not impact CDF, the relevant criterion is LERF. The increase in LERF resulting from a change in the Type A ILRT test interval from 3-per-10 years to once-per-16.25 years is 1.54E-7/yr. Guidance in Reg. Guide 1.174 defines small changes in LERF as below 10^{-6}/yr. Because this delta LERF falls within NRC RG 1.174 Region II ("Small Changes" in risk), the total TMI-1 LERF (3.53E-06/yr) was estimated and shown to be less than the RG 1.174 limit of 1E-5/yr. Therefore, increasing the TMI-1 ILRT interval from the baseline 3-per-10 years to once in 16.25 years results in a 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 $\Delta CCFP_{\%}$ is found to be very small and represents a negligible change in the TMI-1 defense-in-depth. The cumulative impact due to increasing the test frequency from 3-per-10 years to once-per-16.25 years was calculated to be 1.0%.

The change in population dose is also consistent with previously approved ILRT interval extension requests. The incremental change in population dose rate from the current one-time 15-year ILRT extension to 16.25 years is insignificant with an increase of only 0.018 person-rem/yr. The cumulative change from the baseline 3-per-10 year frequency to the one-time 16.25-year extension is also very small, being an increase of 0.188 person-rem/yr.

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 is a real risk impact associated with the setup and performance of the ILRT during shutdown operation [8]. While these risks have not been quantified for TMI-1, 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

The impact of external events on this ILRT risk assessment is summarized in this section (refer to Appendix A for further detail).

Given the characteristics of this 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 events.

Although it is not possible at this time to incorporate realistic quantitative risk assessments of all external event hazards into this assessment (i.e., the spectrum of external hazards have been evaluated in the TMI-1 IPEEE to varying levels of screening and conservatism), the quantitative results of the TMI-1 IPEEE have been evaluated as a sensitivity case to show that the conclusions of this analysis would not be altered if external events were explicitly considered.

The quantitative consideration of external hazards is discussed in more detail in Appendix A of this report. The assessment of the external events uses the results of the TMI-1 IPEEE and does not modify the IPEEE analysis and frequency results, but maintains the conservative nature of the risk results. As can be seen from Appendix A, if the conservative results of the TMI-1 IPEEE are used directly in this assessment, the change in LERF yields a delta value of 6.19E-8/yr for an increase in the testing interval from 15 to

16.25 years. This delta LERF falls within NRC RG 1.174 Region III ("Very Small Changes" in risk). For the internal and external cumulative risk impact of increasing the ILRT frequency from 3-per-10 years to once-in-16.25 years, the delta LERF was 6.56E-7/yr ("Small Changes" in risk). Since this value is greater than 1E-7/yr, the RG 1.174 requirement that the total LERF be less than 1E-5/yr was satisfied by the TMI-1 LERF (internal plus external events) value of 5.58E-6/yr. Therefore, increasing the TMI-1 ILRT interval from the baseline 3-per-10 years to once in 16.25 years results in a small change in risk, and remains an acceptable plant change from a risk perspective when considering external events.

Therefore, incorporating the conservative TMI-1 IPEEE external events accident sequence results into this analysis does not change the conclusion of this risk assessment (i.e., increasing the TMI-1 ILRT interval from once-in-15 to once-in-16.25 years is an acceptable plant change from a risk perspective).

5.4 PARAMETRIC SENSITIVITY ANALYSIS

A parametric sensitivity study was performed in Appendix B that took the PDS frequencies for internal events listed in Table 2-2 and imposed a lognormal distribution with their base frequencies equal to a mean value and a 95th percentile value assumed to be three times the mean. An Excel spreadsheet model was created using Crystal Ball 2000 (ver. 5.1) standard edition software to simulate 20000 iterations using a Latin Hypercube sampling routine. The lognormal distribution values for each of the PDS frequencies are represented in Table B-1 and the distributions for each of the outputs of interest are displayed in Table B-2. Figures B-1 through B-6 graphically display the variability of results.

The results of this sensitivity study showed that even at the 95th percentile value for delta LERF, the acceptance criteria of RG 1.174 was still satisfied. As can be seen from Table B-2 of Appendix B, the 95th percentile for delta LERF was 3.03E-8 for an increase in the testing interval from 15 to 16.25 years. This delta LERF falls within NRC RG 1.174 Region III ("Very Small Changes" in risk). For the internal events cumulative risk impact of increasing the ILRT frequency from 3-per-10 years to once-in-16.25 years, the delta LERF was 3.21E-7/yr ("Small Changes" in risk) at the 95th percentile. Since this value is greater than 1E-7/yr, the RG 1.174 requirement that the total LERF be less than 1E-5/yr was satisfied by the 95th percentile TMI-1 LERF value of 7.29E-6/yr. Therefore, increasing the TMI-1 ILRT interval from the baseline 3-per-10 years to once in 16.25 years results in a small change in risk, even at the 95th percentile, and remains an acceptable plant change from a risk perspective when considering this sensitivity analysis.

5.5 CORROSION SENSITIVITY ANALYSIS

A sensitivity study was performed in Appendix C that was similar to that performed for the previous ILRT extension [32]. The analysis was meant to determine the change in likelihood, due to extending the ILRT inspection frequency, of detecting corrosion of the containment steel shell. This likelihood is then used to determine the resulting change in risk. The results of that analysis are presented in Appendix C.

5.6 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.

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

Section 6

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Appendix A
EXTERNAL EVENT ASSESSMENT

A.1 INTRODUCTION

This appendix discusses the external events assessment in support of the TMI-1 ILRT frequency extension risk assessment. Since a similar process was followed as was done for the treatment of internal events described in Section 3, the same spreadsheet was used to aid in the calculation and tabulation of results [34].

External hazards were evaluated in the TMI-1 Individual Plant Examination of External Events (IPEEE) Submittal in response to the NRC IPEEE Program. The IPEEE Program was a one-time review of external hazard risk to identify potential plant vulnerabilities and to understand severe accident risks. TMI-1 does not currently maintain external event PRA models and associated documentation. Although the external event hazards in the TMI-1 IPEEE were evaluated to varying levels of conservatism, the results of the TMI-1 IPEEE are nonetheless used in this risk assessment to provide a comparative understanding of the impact of external hazards on the conclusions of this ILRT interval extension risk assessment.

The TMI-1 IPEEE study evaluated the following categories of external hazards:

- Seismic Events
- Internal Fires
- External Floods
- High Winds
- Other (e.g., aircraft impacts, nearby facility hazards, etc.)

Consistent with NRC guidance for the IPEEE Program (NUREG-1407), TMI-1 employed probabilistic screening approaches to screen out many hazards from unnecessary detailed analysis, and analyzed the more significant hazards with further probabilistic analysis.

The TMI-1 IPEEE results are summarized in Table A-1. As can be seen from Table A-1, seismic events, internal fires, and external floods contribute 99% to the plant risk resulting from external hazards. As such, these three specific hazards are reviewed as part of this ILRT risk assessment, and the other external event hazards are reasonably assumed not to impact the results or conclusions of this risk assessment.

The seismic event, internal fires and external flood analyses of the TMI-1 IPEEE are summarized below.

A.2 TMI-1 IPEEE SEISMIC ANALYSIS

Seismic event hazards at TMI-1 were evaluated in the TMI-1 IPEEE using a seismic PRA approach (i.e., as opposed to the deterministic-based seismic margins approach available as an alternative methodology for many licensees). As such, although the analysis is conservative, quantitative insights regarding the relative frequency and associated characteristics of seismic-induced severe accidents is available from the TMI-1 IPEEE.

The seismic-induced core damage frequency results from the TMI-1 IPEEE, as a function of Plant Damage State (PDS), are summarized in Table A-2. As can be seen from Table A-2, over 75% of the seismic accident sequences result in PDS7F (small LOCA, no injection, steam generators available, no containment safeguards available, containment isolated).

The TMI-1 IPEEE does not analyze explicitly release category information. As such, a breakdown of release category (RC) frequency results for seismic events is not available from the TMI-1 IPEEE. However, PDS vs RC relationship information available from the TMI-1 internal events PRA can be used to provide a reasonable representation of the spectrum of seismic-induced radionuclide releases.

Table A-1

RESULTS OF TMI-1 IPEEE
- Contributions to External Event CDF -

External Event	Contribution to External Event CDF (%)
Seismic Events	23.6
Internal Fires	15.8
External Floods	59.6
High Winds	0.6
Aircraft Crash	0.3
Hazardous Chemical Accidents	0.1
Other	negligible

Table A-2

SUMMARY OF TMI-1 IPEEE SEISMIC ANALYSIS RESULTS

PDS	PDS Definition	External Events CDF	
		PDS Frequency (1/yr)	Contribution (%)
PDS7F	Small LOCA, injection failure, steam generators available, no containment safeguards available, containment isolated	2.43E-5	75.6
PDS4F	Medium LOCA, injection failure, no containment safeguards available, containment isolated	6.42E-6	20.0
PDS7R	Small LOCA, injection failure, steam generators available, no containment safeguards available, large containment isolation failure	5.47E-7	1.7
PDS4C	Medium LOCA, injection failure, fans available, sprays unavailable in injection and recirculation modes, containment isolated	1.86E-7	0.58
PDS7L	Small LOCA, injection failure, steam generators available, no containment safeguards available, small containment isolation failure	1.77E-7	0.55
PDS4R	Medium LOCA, injection failure, no containment safeguards available, large containment isolation failure	1.46E-7	0.45
PDS1F	Large LOCA, injection failure, no containment safeguards available, containment isolated	1.43E-7	0.44
PDS4E	Medium LOCA, injection failure, sprays in injection mode available, fans unavailable, sprays unavailable in recirculation mode, containment isolated	9.47E-8	0.29
All Others	All other PDS categories.	1.25E-7	0.39
	TOTAL:	3.21E-5	100

Using the PDS vs RC ratios from the TMI-1 internal events Level 2 PRA information provided in Table 2-3 of this report, release category frequencies representative of the TMI-1 IPEEE seismic analysis are summarized in Table A-3. This information is used in Section A.5 of this appendix to provide insight into the impact of external hazard risk on the conclusions of this ILRT risk assessment.

A.3 TMI-1 IPEEE INTERNAL FIRES ANALYSIS

Internal fire hazards at TMI-1 were evaluated in the TMI-1 IPEEE using a probabilistic screening approach based on the EPRI Fire Induced Vulnerability Evaluation (FIVE) methodology. As such, although the analysis is conservative, quantitative insights regarding the relative frequency and associated characteristics of fire-induced severe accidents is available from the TMI-1 IPEEE.

The core damage frequency results from the TMI-1 IPEEE for the unscreened fire areas, as a function of Plant Damage State (PDS), are summarized in Table A-4. As can be seen from Table A-4, over 50% of the fire accident sequences result in PDS7F (small LOCA, no injection, steam generators available, no containment safeguards available, containment isolated).

The TMI-1 IPEEE does not analyze explicitly release category information. As such, a breakdown of release category (RC) frequency results for fire events is not available from the TMI-1 IPEEE. However, PDS vs RC relationship information available from the TMI-1 internal events PRA can be used to provide a reasonable representation of the spectrum of fire-induced radionuclide releases. This information is also summarized in Table A-4.

Table A-3

PDS AND ESTIMATED RELEASE CATEGORY FREQUENCIES FOR SEISMIC-INDUCED ACCIDENTS

TMI-1 IPEEE Seismic Results		Estimated Release Category Frequencies (1) (1/yr)							
PDS	PDS Frequency (1/yr)	RC306	RC502	RC605	RC606	RC703	RC704	RC801	RC903
PDS4F	6.42E-6	---	6.42E-8	1.28E-7	---	1.28E-6	---	2.50E-6	2.44E-6
PDS7F	2.43E-5	---	2.43E-7	2.43E-7	1.94E-6	1.94E-6	1.75E-5	2.43E-6	---
PDS7R (2) (3)	1.38E-6	1.38E-6	---	---	---	---	---	---	---
TOTALS:	3.21E-5	1.38E-6	3.07E-7	3.71E-7	1.94E-6	3.23E-6	1.75E-5	4.93E-6	2.44E-6

NOTES:

(1) Seismic Release Category frequencies obtained by multiplying the seismic PDS frequency from the TMI-1 IPEEE by the TMI-1 internal events PRA PDS-to-RC ratios obtained from the information in Table 2-3 of this report. Only those Release Categories with non-zero frequencies are summarized in this table.

(2) For simplicity, all the negligible seismic contributors summarized in Table A-2 are summed into one PDS (i.e., PDS7R) for this risk assessment.

(3) Release Category frequency information for the PDS7R Plant Damage State is not available from the TMI-1 PRA. Based on the definitions of PDS7R and RC306 (refer to Tables 2-2 and 2-3 of this report for these definitions), the entire seismic CDF frequency of PDS7R is reasonably assigned to Release Category RC306.

Table A-4

PDS AND ESTIMATED RELEASE CATEGORY FREQUENCIES FOR FIRE-INDUCED ACCIDENTS

TMI-1 IPEEE Internal Fire Results		Estimated Release Category Frequencies (1) (1/yr)								
PDS	PDS Frequency (1/yr)	RC501	RC502	RC605	RC606	RC703	RC704	RC801	RC901	RC903
PDS4C (2)	3.24E-6	---	6.48E-8	---	2.92E-7	---	2.66E-6	2.27E-7	---	---
PDS4F	1.73E-6	---	1.73E-8	3.46E-8	---	3.46E-7	---	6.74E-7	---	6.57E-7
PDS7C	1.30E-6	---	2.59E-8	---	1.04E-7	5.18E-8	9.72E-7	1.04E-7	---	3.89E-8
PDS7E	1.51E-6	---	3.02E-8	---	1.36E-7	---	1.18E-6	1.66E-7	---	---
PDS7F	1.12E-5	---	1.12E-7	1.12E-7	8.99E-7	8.99E-7	8.09E-6	1.12E-6	---	---
PDS8A	1.51E-6	9.07E-8	---	---	---	---	---	7.56E-8	1.35E-6	---
PDS8E	1.08E-6	---	2.16E-8	---	9.72E-8	---	8.42E-7	1.19E-7	---	---
TOTALS:	2.16E-5	9.07E-8	2.72E-7	1.47E-7	1.53E-6	1.30E-6	1.37E-5	2.49E-6	1.35E-6	6.96E-7

NOTES TO TABLE A-4:

(1) Fire Release Category frequencies obtained by multiplying the fire PDS frequency from the TMI-1 IPEEE by the TMI-1 internal events PRA PDS-to-RC ratios obtained from the information in Table 2-3 of this report. Only those Release Categories with non-zero frequencies are summarized in this table.

(2) For simplicity, all the minor fire contributors are summed into one PDS (i.e., PDS4C) for the purposes of this risk assessment. This is the most conservative PDS assignment as it translates into the largest contribution to the delta LERF calculation.

A.4 TMI-1 IPEEE EXTERNAL FLOODS ANALYSIS

External flooding hazards at TMI-1 were evaluated in the TMI-1 IPEEE using probabilistic accident sequence analysis. As such, although the analysis is conservative, quantitative insights regarding the relative frequency and associated characteristics of external flooding severe accidents is available from the TMI-1 IPEEE.

The external flooding core damage frequency results from the TMI-1 IPEEE are summarized in Table A-5. Various individual external flooding accident sequences are evaluated in the TMI-1 IPEEE external flooding analysis, the results in Table A-5 present the summed frequency results as a function of external flood initiator. As can be seen from Table A-5, the TMI-1 IPEEE external flooding analysis evaluated three general categories of external floods:

1. External flood elevations below Elevation 305' (site will not be impacted unless dike fails)
2. External flood elevations above Elevation 310' (critical plant structures will be flooded despite implementation of flood protective measures per plant procedures)
3. External flood elevations between Elevations 305' – 310'

The TMI-1 IPEEE external flooding analysis does not explicitly assign Plant Damage State categories to the analyzed external flooding core damage sequences. However, accident sequence descriptions provided in the TMI-1 IPEEE documentation provide sufficient information in most cases to allow PDS categories to be assigned. Using the accident sequence description information in the TMI-1 IPEEE and the TMI-1 PDS definitions summarized in Table 2-2 of this report, the TMI-1 IPEEE external flooding results are summarized in Table A-6 as a function of PDS.

Table A-5

SUMMARY OF TMI-1 IPEEE EXTERNAL FLOODING ANALYSIS RESULTS
(External Flooding CDF as a Function of External Flooding Initiator)

External Flooding Initiator	Initiator Frequency (1/yr)	External Flooding CDF	
		Frequency (1/yr)	Contribution (%)
External flood elevations below Elevation 305' (site will not be impacted unless dike fails)	2.50E-3	2.50E-7	0.3
External flood elevations above Elevation 310' (critical plant structures will be flooded despite implementation of flood protective measures per plant procedures)	2.50E-4	6.37E-5	78.6
External flood elevations between Elevations 305' – 310'	4.50E-4	1.71E-5	21.1
	TOTAL:	8.10E-5	100

Table A-6

ESTIMATED RELEASE CATEGORY FREQUENCIES FOR EXTERNAL FLOODING ACCIDENTS

TMI-1 IPEEE Ext. Flooding Results		Estimated Release Category Frequencies [(1)] (1/yr)										
PDS	PDS Frequency (1/yr)	RC102	RC104	RC501	RC502	RC605	RC606	RC703	RC704	RC801	RC901	RC903
PDS5A	6.36E-6			3.82E-7						3.18E-7	5.66E-6	
PDS7A	8.65E-8			5.19E-9						4.33E-9	7.70E-8	
PDS7C	1.72E-5				3.44E-7		1.38E-6	6.88E-7	1.29E-5	1.38E-6		5.16E-7
PDS7F	3.95E-5				3.95E-7	3.95E-7	3.16E-6	3.16E-6	2.84E-5	3.95E-6		
PDS10A	1.79E-5	1.49E-5	7.16E-7	1.79E-7						1.79E-7	1.97E-6	
TOTALS:	8.10E-5	1.49E-5	7.16E-7	5.66E-7	7.39E-7	3.95E-7	4.54E-6	3.85E-6	4.13E-5	5.83E-6	7.71E-6	5.16E-7

NOTES:

(1) External Flooding Release Category frequencies obtained by multiplying the external flooding PDS frequency by the TMI-1 internal events PRA PDS-to-RC ratios obtained from the information in Table 2-3 of this report. Only those Release Categories with non-zero frequencies are summarized in this table.

(2) Based on the accident sequence description information in the TMI-1 IPEEE and the TMI-1 PDS definitions summarized in Table 2-2 of this report, the assignment of the external flooding accident sequences to PDS category is performed as follows:

- >310' El. Floods: 56% of CDF (of sequences due to this IE) assigned to PDS7F; 28% to PDS10A; and 16% to PDS7C
- <305' El. Floods: 56% of CDF (of sequences due to this IE) assigned to PDS7F; 28% to PDS10A; and 16% to PDS7C
- 305' – 310' El. Flood: Seqs. A & B assigned to PDS5A, seqs. C & D to PDS7C, seq. E to PDS7F, and seq. F to PDS7A.

Table A-6 also provides a breakdown of external flooding PDS versus release category. The TMI-1 IPEEE does not analyze explicitly release category information. As such, a breakdown of release category (RC) frequency results for external flooding events is not available from the TMI-1 IPEEE. However, PDS vs RC relationship information available from the TMI-1 internal events PRA can be used to provide a reasonable representation of the spectrum of seismic-induced radionuclide releases. Using the PDS vs RC ratios from the TMI-1 internal events Level 2 PRA information provided in Table 2-3 of this report, release category frequencies representative of the TMI-1 IPEEE external flooding analysis are summarized in Table A-6. This information is used in Section A.5 of this appendix to provide insight into the impact of external hazard risk on the conclusions of this ILRT risk assessment.

A.5 IMPACT OF EXTERNAL HAZARD RISK ON ILRT RISK ASSESSMENT

The NEI Interim Guidance methodology performed in Section 3 of this report is re-performed here including, in addition to internal event information, the TMI-1 IPEEE based external event release category information discussed in the previous sections.

A.5.1 Baseline EPRI Category Frequencies (Including TMI-1 External Events)

The baseline EPRI category frequencies are estimated here in the same manner as that described in Section 3 of this report, except that the TMI-1 IPEEE based estimates of external event initiated release category contributions are included.

Frequency of EPRI Category 1

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 RC9 in the TMI-1 Level 2 PRA).

The frequency of TMI-1 severe accidents in which the containment remains intact is 1.28E-5/yr (due to internal events) + 1.27E-5/yr (due to external events) = 2.55E-5/yr. As described below, the frequencies of the 3a and 3b categories are 2.14E-6/yr and 2.14E-7/yr, respectively. Therefore, the frequency of EPRI Category 1 is calculated as (2.55E-5/yr) – (2.14E-6/yr + 2.14E-7/yr) = 2.32E-5/yr.

Frequency of EPRI Category 2

The frequency of this EPRI category is determined by summing the frequencies of the RC3-series and RC4-series release categories. As discussed in the notes to Table 2-3, the RC3-series and the RC4-series release categories represent severe accidents with containment isolation failure. The internal events contribution is 5.19E-8/yr (refer to Section 3.1). The external events contribution is 1.38E-6/yr. Therefore, the frequency of EPRI Category 2 is 5.19E-8/yr + 1.38E-6/yr = 1.43E-6/yr.

Frequency of EPRI Category 3a

As discussed in more detail in Section 3.1, the frequency per year for this category is calculated as:

Frequency 3a = [3a conditional failure probability] x [CDF - (CDF with containment failure independent of containment leakage)]

Also as discussed in Section 3.1, EPRI Category 3a is refined in this risk assessment into accidents with containment sprays available (3a Scrubbed) and with sprays unavailable (3a Unscrubbed).

In addition to the above, the refinement (consistent with supplemental NEI guidance) is made for external event contributors to exclude External Flooding scenarios from the frequency calculation of category 3a. Per TMI-1 Emergency Procedure 1202-32, Flood, a plant shutdown will be initiated at a river elevation of 302' El., approximately 8 feet below the TMI-1design basis external flooding elevation (310' El.). Per the TMI-1 design basis Probable Maximum Flood characteristics, plant shutdown would be initiated about 27 hours prior to flood height reaching an elevation of 310 feet. In the TMI-1 IPEEE a worst-case hurricane was also considered, and in this case the design basis flood elevation would be exceeded in approximately 5 hours after shutdown was initiated. As such, given the slow-developing nature of external flooding scenarios, such scenarios would not result in LERF releases because of the delayed time to core damage.

Therefore, the frequency of category 3a (Scrubbed) is calculated as (2.70E-02) x (1.44E-05/yr [internal events contribution] + 9.55E-06/yr [external events contribution]) = 6.46E-07/yr. Likewise, the frequency of category 3a (Unscrubbed) is calculated as (2.70E-02) x (1.29E-05/yr [internal events contribution] + 4.21E-05/yr [external events contribution]) = 1.49E-06/yr.

<u>Frequency of EPRI Category 3b</u>

In the same manner as that discussed previously for category 3a, EPRI Category 3b is refined into accidents with containment sprays available (3b Scrubbed) and with sprays unavailable (3b Unscrubbed). The frequencies are calculated in the same manner as that described above for 3a, except that the 3b pre-existing leakage probability is 2.7E-3.

Therefore, the frequency of category 3b (Scrubbed) is calculated as (2.70E-03) x (1.44E-05/yr [internal events contribution] + 9.55E-06/yr [external events contribution]) = 6.46E-08/yr. Likewise, the frequency of category 3b (Unscrubbed) is calculated as (2.70E-03) x (1.29E-05/yr [internal events contribution] + 4.21E-05/yr [external events contribution]) = 1.49E-07/yr.

Frequency of EPRI Category 4

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

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

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). As discussed in Section 3.1, the frequency of this category is calculated by summing the frequencies of release categories RC5, RC6, RC7, and RC8. The internal events contribution is 1.53E-5/yr (refer to Section 3.1). The external events contribution is 1.05E-4/yr. Therefore, the frequency of EPRI Category 7 is 1.57E-5/yr + 1.05E-4/yr = 1.21E-4/yr.

Frequency of EPRI Category 8

As discussed in Section 3.1, the frequency of EPRI category 8 is calculated by summing the frequencies of release categories RC1 and RC2. The internal events contribution is 3.41E-6/yr (refer to Section 3.1). The external events contribution is 1.56E-5/yr. Therefore, the frequency of EPRI Category 8 is 3.44E-6/yr + 1.56E-5/yr = 1.90E-5/yr.

A.5.2 Dose Rate Estimates (Including TMI-1 External Events)

The baseline dose as a function of EPRI category are estimated in the same manner as that discussed in Section 3.3.1. As the doses are calculated on a weighted average contribution to dose rate basis, the doses estimated when external event accident frequencies are included vary slightly from that presented in the base analysis (internal events only) in Section 3. The baseline doses and dose rates (incorporating external events) as a function of EPRI category are summarized in Table A-7.

A.5.3 Change in LERF (Including TMI-1 External Events)

As discussed in Section 3.4.4, the change in LERF associated with extending the ILRT interval is determined by the change in the frequency of EPRI category 3b (Unscrubbed). As can be seen from Table A-7, the increase in the LERF risk measure due to extending the ILRT from 1-per-15 years to 1-per-16.25 years is 6.19E-8/yr.

Comparison to RG 1.174 Acceptance Guidelines

NRC Regulatory Guide 1.174, "An Approach for Using PRA in Risk-Informed Decisions on Plant-Specific Changes to the Licensing Basis", provides NRC recommendations for using risk information in support of applications requesting changes to the license basis of

Table A-7

QUANTITATIVE RESULTS AS A FUNCTION OF ILRT INTERVAL (INCLUDING EXTERNAL EVENTS)

EPRI Category	Dose (Person-Rem Within 50 miles)	Quantitative Results as a Function of ILRT Interval					
		Methodology Baseline (3-per-10 year ILRT)		Current (1-per-15 year ILRT)		Proposed (1-per-16.25 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	6.12E+02	2.32E-05	1.42E-02	1.38E-05	8.42E-03	1.28E-05	7.82E-03
2	3.91E+06	1.43E-06	5.60E+00	1.43E-06	5.60E+00	1.43E-06	5.60E+00
3a (Scrubbed)	6.07E+03	6.46E-07	3.92E-03	3.23E-06	1.96E-02	3.50E-06	2.12E-02
3a (Unscrubbed)	8.69E+04	1.49E-06	1.29E-01	7.43E-06	6.45E-01	8.05E-06	6.99E-01
3b (Scrubbed)	2.12E+04	6.46E-08	1.37E-03	3.23E-07	6.86E-03	3.50E-07	7.44E-03
3b (Unscrubbed)	3.04E+05	1.49E-07	4.52E-02	7.43E-07	2.26E-01	8.05E-07	2.45E-01
4	N/A	N/A	N/A	NA	N/A	NA	N/A
5	N/A	N/A	N/A	NA	N/A	NA	N/A
6	N/A	N/A	N/A	NA	N/A	NA	N/A
7	4.04E+05	1.21E-04	4.88E+01	1.21E-04	4.88E+01	1.21E-04	4.88E+01
8	1.02E+06	1.90E-05	1.94E+01	1.90E-05	1.94E+01	1.90E-05	1.94E+01
TOTALS:		1.67E-04	74.01	1.67E-04	74.72	1.67E-04	74.80
Increase in Dose Rate					7.1E-01		7.4E-02
Increase in LERF				5.94E-07		6.19E-08	
Increase in $CCFP_{\%}$				0.5		0.1	

the plant. As discussed in Section 2 of this report, the risk acceptance criteria of RG 1.174 is used here to assess the ILRT interval extension.

The 6.19E-8/yr increase in LERF from extending the TMI-1 ILRT frequency from 1-per-15 years to 1-per-16.25 years falls into Region III ("Very Small Change" in risk) of the RG 1.174 acceptance guidelines and represents an incremental change in risk. The cumulative change in risk in increasing the ILRT testing frequency from 3-per-10 years to once-in-16.25 years is 6.56E-7/yr. Per RG 1.174, when the calculated increase in LERF due to the proposed plant change is in the range of 1E-7 to 1E-6 per reactor year, the risk assessment must also reasonably show that the total LERF is less than 1E-5/yr.

As discussed in the TMI-1 PRA documentation, the following TMI-1 PRA release categories contribute to the LERF risk measure:

- RC102
- RC104
- RC202
- RC204
- RC302
- RC304
- RC305
- RC306
- RC402
- RC404
- RC406
- RC408
- RC501
- RC502

Comparison of the above list to the TMI-1 Level 2 internal events PRA results summarized in Table 2-3 of this report, the LERF contribution from internal events is estimated at 3.53E-6/yr. Using the TMI-1 IPEEE results, the LERF contribution from external events is estimated at 2.05E-6/yr (as discussed earlier, External Floods are excluded from consideration as LERF contributors). Therefore, the total LERF for TMI-1 is estimated at 3.53E-6/yr + 2.05E-6/yr = 5.58E-6/yr, which is less than the RG 1.174 limit of 1E-5/yr.

Appendix B
Parametric Sensitivity Analysis

B.1 INTRODUCTION

This appendix discusses the sensitivity study that was performed to gain insight on the variability of the results for changes in dose, $CCFP_{\%}$, and LERF as a result of increasing the ILRT frequency from once in 15 years to once in 16.25 years.

Each of the plant damage state frequencies listed in Table 2-2 were given a lognormal distribution, with the base frequency representing a mean value. A 95th percentile value was assumed to be a factor of three higher than the mean. The lognormal distribution values for each of the PDS frequencies are represented in Table B-1. An Excel spreadsheet model was created using Crystal Ball 2000 (ver. 5.1) standard edition software to simulate 20000 iterations using a Latin Hypercube sampling routine [35]. The distributions for each of the outputs of interest are displayed in Table B-2. Figures B-1 through B-6 graphically display the resultant distribution for each of the results.

B.2 SUMMARY

In summary, even at the 95th percentile, the criteria for delta LERF still satisfies the RG 1.174 criteria for cumulative risk in changing the testing frequency from the baseline 3-per-10 years to once-in-16.25 years. Therefore, this sensitivity analysis does not change the overall conclusion of this risk assessment (i.e., increasing the TMI-1 ILRT interval from the current once-in-15 to once-in-16.25 years is an acceptable plant change from a risk perspective).

Table B-1

PLANT DAMAGE STATE LOGNORMAL FREQUENCY DISTRIBUTIONS

PDS Designator	Mean	5th Percentile	95th Percentile
PDS10A	1.45E-06	4.82E-07	4.34E-06
PDS10C	1.04E-07	3.47E-08	3.12E-07
PDS12C	1.40E-08	4.67E-09	4.20E-08
PDS15A	1.30E-07	4.35E-08	3.91E-07
PDS18B	0.00E+00	0.00E+00	0.00E+00
PDS18C	1.74E-06	5.79E-07	5.21E-06
PDS18E	0.00E+00	0.00E+00	0.00E+00
PDS18F	2.22E-08	7.38E-09	6.65E-08
PDS19O	1.83E-07	6.09E-08	5.48E-07
PDS1A	9.35E-08	3.12E-08	2.81E-07
PDS1C	1.21E-09	4.04E-10	3.64E-09
PDS2B	6.85E-08	2.28E-08	2.06E-07
PDS3A	2.43E-09	8.10E-10	7.29E-09
PDS4A	1.02E-08	3.40E-09	3.06E-08
PDS4B	1.05E-08	3.51E-09	3.16E-08
PDS4C	2.01E-09	6.71E-10	6.04E-09
PDS4F	1.85E-09	6.18E-10	5.56E-09
PDS4L	0.00E+00	0.00E+00	0.00E+00
PDS5A	1.62E-07	5.40E-08	4.86E-07
PDS5B	1.17E-08	3.91E-09	3.52E-08
PDS5C	1.56E-08	5.19E-09	4.67E-08
PDS5F	9.51E-07	3.17E-07	2.85E-06
PDS6A	1.42E-08	4.73E-09	4.26E-08
PDS7A	5.43E-06	1.81E-06	1.63E-05
PDS7C	3.07E-06	1.02E-06	9.21E-06
PDS7D	1.24E-07	4.12E-08	3.71E-07
PDS7E	3.27E-10	1.09E-10	9.81E-10
PDS7F	2.63E-06	8.75E-07	7.88E-06
PDS7L	5.25E-08	1.75E-08	1.57E-07
PDS8A	8.20E-06	2.73E-06	2.46E-05
PDS8B	2.89E-06	9.64E-07	8.68E-06
PDS8C	4.45E-06	1.48E-06	1.34E-05
PDS8D	8.44E-08	2.81E-08	2.53E-07
PDS8E	3.00E-08	1.00E-08	9.00E-08

Table B-2

OUTPUT SUMMARY FOR PARAMETERS OF INTEREST

Parameter	Mean	5^{th} Percentile	95^{th} Percentile
$\Delta LERF_{(16.25\text{-}Base)}$ (1/yr)	1.54E-07	1.05E-07	3.21E-07
$\Delta LERF_{(16.25\text{-}15)}$ (1/yr)	1.45E-08	9.94E-09	3.03E-08
$\Delta CCFP_{\%(16.25\text{-}Base)}$	1.02%	0.90%	1.09%
$\Delta CCFP_{\%(16.25\text{-}15)}$	0.10%	0.08%	0.10%
$Dose_{16.25\text{-}Base}$ (person-rem/yr)	0.188	0.132	0.388
$Dose_{16.25\text{-}15}$ (person-rem/yr)	0.018	0.012	0.037
LERF for Internal Events (1/yr)	3.53E-06	2.51E-06	7.29E-06

Figure B-1

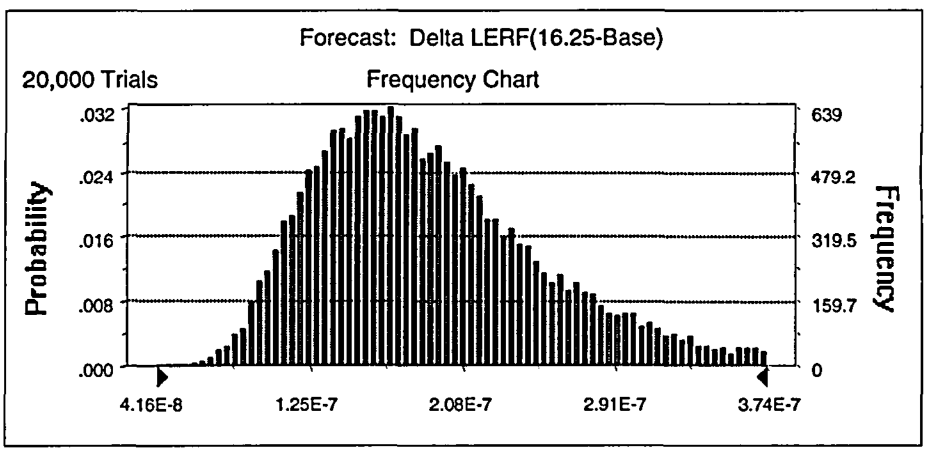


Figure B-2

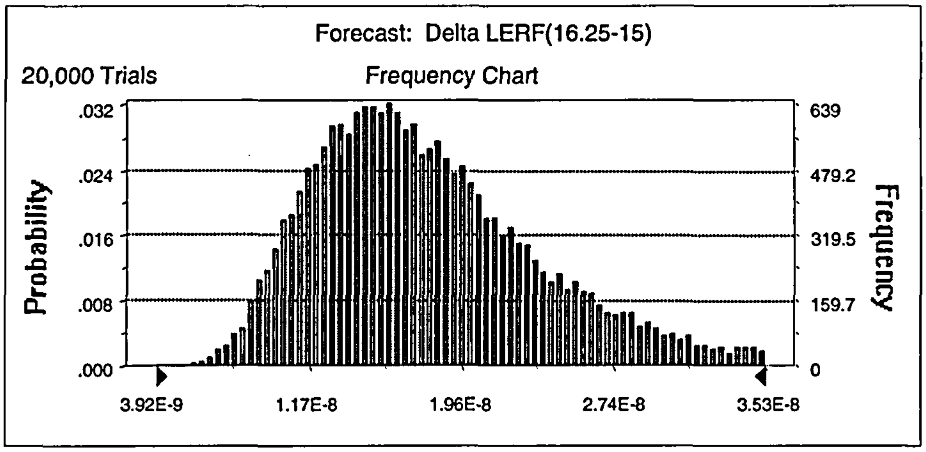


Figure B-3

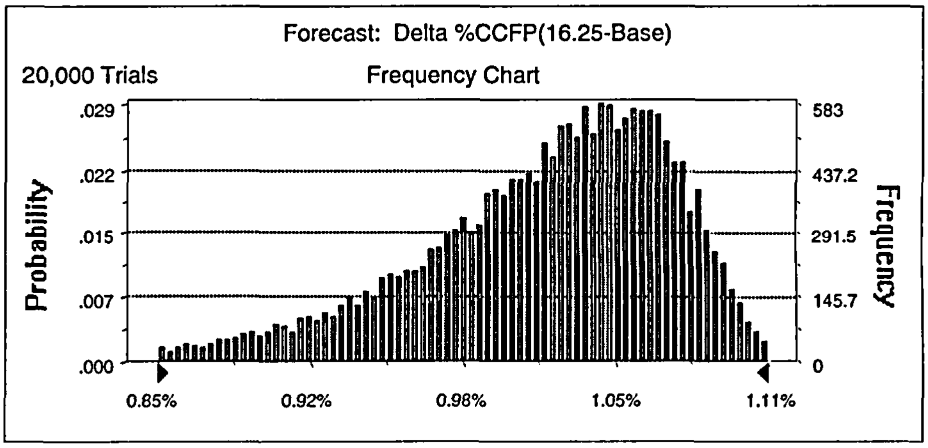


Figure B-4

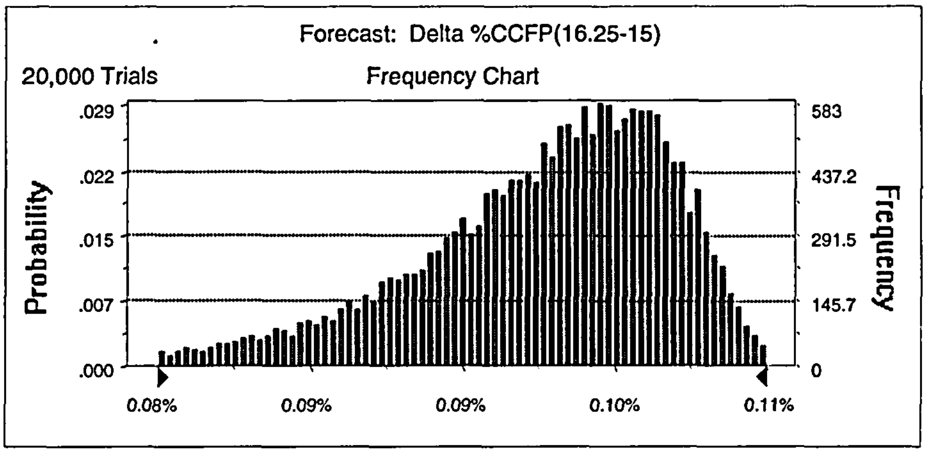


Figure B-5

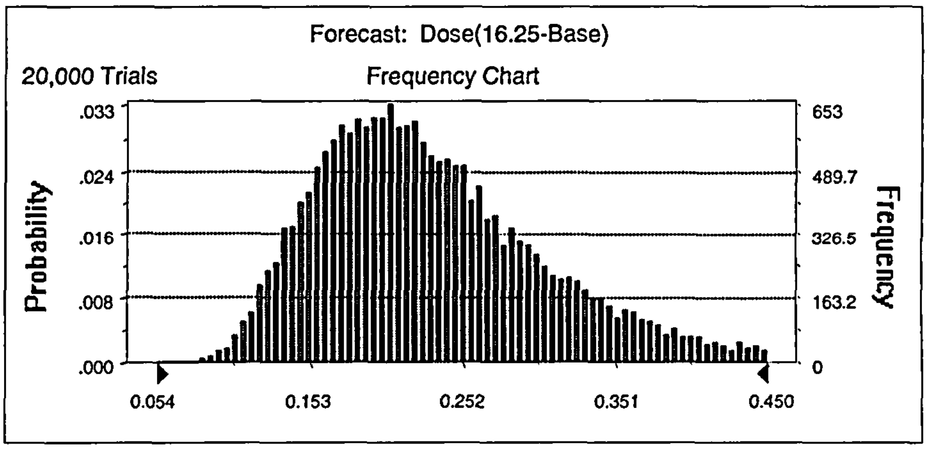


Figure B-6

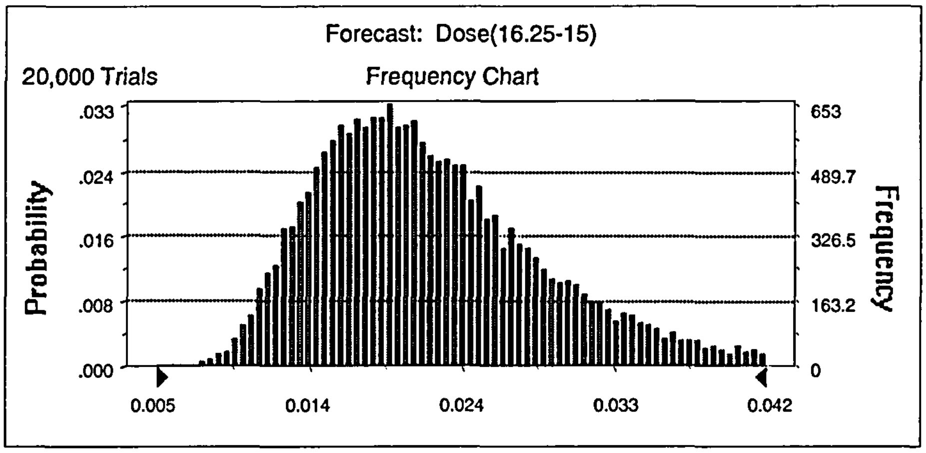
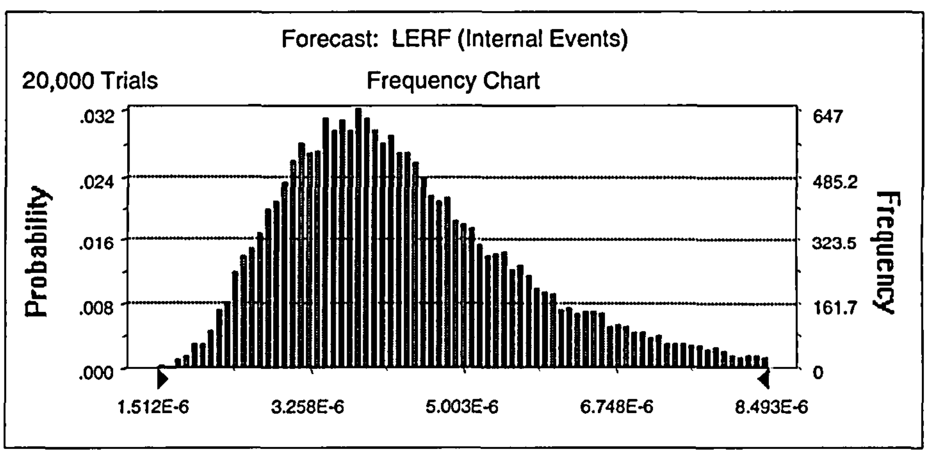


Figure B-7



Appendix C
Corrosion Sensitivity Analysis

C.1 DISCUSSION OF RESULTS

The results of considering the effects of steel liner corrosion in extending the ILRT testing frequency to once-in-16.25 years are summarized below in Table C.1. The approach used was identical to that performed earlier for a previous ILRT test interval extension [32], with the exception that the test interval was extended to a 16.25 year period. Also, since this is another sensitivity analysis, only internal event scenarios were considered for steel liner corrosion effects; see Appendix A for a separate treatment of external events.

As can be seen in Table C.1, with base case assumptions, the impact from including the effects of corrosion is very minimal. The potential effect of corrosion increases the estimated delta-LERF from 1.5E-7 to 1.7E-7 for an ILRT interval extension from 3-per-10 years to once-in-16.25 years. Other sensitivity cases were performed by adjusting such parameters as the corrosion rate and likelihood of detection failure. Even the upper bound estimates with conservative assumptions for all of the key parameters would yield a delta-LERF value of less than 1E-6. Hence, these results confirm the conclusion that adjusting the ILRT test frequency from a baseline 3-per-10 years to once-in-16.25 years results in a small change in cumulative risk, consistent with the criteria found in Regulatory Guide 1.174, and is an acceptable plant change from a risk perspective.

Table C-1

Corrosion Sensitivity Cases

Corrosion Case	Type of Containment Breach	Visual Inspection & Non-Visual Flaws	Likelihood Flaw is EPRI Class 3b	Increase in Delta LERF Due to Corrosion Effects	Delta LERF for Each Case Due to ILRT Extension from 3 in 10 to once in 16.25 years
Base Case Corrosion Doubles every 5 yrs	**Base Case** (10% Wall, 1% Floor)	**Base Case** 10%	**Base Case** 100%	**Base Case** 1.78E-08	**Base Case** 1.72E-07
Doubles every 2 yrs	Base	Base	Base	4.65E-08	2.01E-07
Doubles every 10 yrs	Base	Base	Base	1.34E-08	1.67E-07
Base	(100% Wall, 10% Floor)	Base	Base	1.78E-07	3.32E-07
Base	(1% Wall, 0.1% Floor)	Base	Base	1.78E-09	1.56E-07
Base	Base	*15%*	Base	2.49E-08	1.79E-07
Base	Base	*5%*	Base	1.07E-08	1.65E-07
<u>Lower Bound</u>					
Doubles every 10 yrs	(1% Wall, 0.1% Floor)	5%	Base	8.01E-10	1.55E-07
<u>Upper Bound</u>					
Doubles every 2 yrs	(100% Wall, 10% Floor)	15%	Base	6.51E-07	8.05E-07