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10 CFR 50.90

February 20, 2007

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

Limerick Generating Station, Units 1 and 2
Facility Operating License Nos. NPF-39 and NPF-85
<u>NRC Docket Nos. 50-352 and 50-353</u>

Subject: Technical Specifications Change Request -Type A Test Extension

Pursuant to 10 CFR 50.90, Exelon Generation Company, LLC (EGC) hereby requests an amendment to Appendix A, Technical Specifications, of Facility Operating License Nos. NPF-39 and NPF-85. The proposed change modifies Technical Specifications (TS) 6.8.4.g, "Primary Containment Leakage Rate Testing Program." Specifically, the proposed change will revise TS 6.8.4.g to reflect a one-time extension of the containment Type A Integrated Leak Rate Test (ILRT) from 10 to 15 years. This one-time extension will require the Type A ILRT to be performed no later than May 15, 2013, (Unit 1) and May 21, 2014, (Unit 2).

EGC requests approval of the proposed changes by February 20, 2008. 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 20th of February 2007.

Respectfully,

Pamela B. Cowan

Pamela B. Cowan
Director, Licensing & Regulatory Affairs
Exelon Generation Company, LLC

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 LGS, Units 1 and 2 To Support ILRT (Type A) Interval Extension Request

cc: R. R. Janati, Commonwealth of Pennsylvania
S. J. Collins, Administrator, Region 1, USNRC
S. Hansell, USNRC Senior Resident Inspector
J. Shea, Project Manager, USNRC

ATTACHMENT 1

EVALUATION OF PROPOSED CHANGE

ATTACHMENT 1

Evaluation of Proposed Change

CONTENTS

SUBJECT: Type A Test Extension

ATTACHMENT 1
Evaluation of Proposed Change

1.0 INTRODUCTION

This letter is a request to amend Facility Operating License Nos. NPF-39 and NPF-85 for Limerick Generating Station (LGS), Units 1 and 2. The proposed change modifies Technical Specifications (TS) 6.8.4.g, "Primary Containment Leakage Rate Testing Program." Specifically, the proposed change will revise TS 6.8.4.g to reflect a one-time extension of the containment Type A Integrated Leak Rate Test (ILRT) from 10 to 15 years. This one-time extension will require the Type A ILRT to be performed no later than May 15, 2013 (Unit 1) and May 21, 2014 (Unit 2).

Exelon Generation Company, LLC (EGC) requests approval of the proposed change by February 20, 2008. Once approved, the amendment shall be implemented within 60 days.

2.0 PROPOSED CHANGE

The proposed change would revise TS 6.8.4.g ("Primary Containment Leakage Rate Testing Program") of the LGS, Unit 1 Technical Specifications to add the following statement:

> ", as modified by the following exception to NEI 94-01, Rev. 0, "Industry Guideline for Implementing Performance-Based Option of 10 CFR 50, Appendix J":
>
> a. Section 9.2.3: The first Type A test performed after May 15, 1998 shall be performed no later than May 15, 2013."

Additionally, the proposed change would revise TS 6.8.4.g ("Primary Containment Leakage Rate Testing Program") of the LGS, Unit 2 Technical Specifications to add the following statement:

> ", as modified by the following exception to NEI 94-01, Rev. 0, "Industry Guideline for Implementing Performance-Based Option of 10 CFR 50, Appendix J":
>
> a. Section 9.2.3: The first Type A test performed after May 21, 1999 shall be performed no later than May 21, 2014."

3.0 BACKGROUND

The proposed change involves a one-time extension to the ten (10) year frequency of the performance-based leakage rate testing program for Type A tests as required by Nuclear Energy Institute (NEI) 94-01, Revision 0, "Industry Guideline for Implementing Performance-Based Option of 10 CFR Part 50, Appendix J" (Reference 1). The most recent containment Type A Integrated Leak Rate Tests (ILRTs) for LGS, Units 1 and 2 were performed on May 15, 1998 and May 21, 1999, respectively, and would need to be performed no later than Refueling Outage 1R12 (Unit 1) in 2008, and 2R10 (Unit 2) in 2009. The proposed exception would allow the next Type A ILRTs to be performed within fifteen (15) years (i.e., May 15, 2013 (Unit 1) and May 21, 2014 (Unit 2)) from the most recent Type A ILRT as opposed to the current ten (10) year frequency.

This one-time extension will result in the following:

- Perform a Type A ILRT no later than May 15, 2013 (Unit 1) and May 21, 2014 (Unit 2).

- A substantial cost savings will be realized by deferring the Type A test for an additional five (5) years. Cost savings have been estimated for each outage at approximately $1.1 million, which includes labor, equipment and critical path outage time needed to perform the test.

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 Nos. 118 and 81 for LGS, Units 1 and 2 permit implementation of 10 CFR 50, Appendix J, Option B (Reference 2). TS 6.8.4.g 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 3).

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). Based on the ILRT history discussed below, the current test interval is 10 years.

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 LGS, Units 1 and 2 Type A ILRTs demonstrate that the LGS, Units 1 and 2 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 for the LGS, Units 1 and 2 Type A ILRTs are as follows:

a. LGS, Unit 1 ILRT Results

Year	%wt/day
1984	0.1642
1987	0.1469
1990	0.2870
1998	0.3070

b. LGS, Unit 2 ILRT Results

Year	%wt/day
1989	0.2310
1993	0.1836
1999	0.3272

The LGS, Units 1 and 2 TS 6.8.4.g limits the maximum allowable primary containment leakage rate to 0.5% wt/day at P_a (44.0 psig).

4.3 Plant Design and Operational Performance

The containment system, designed by Bechtel Power Corporation, limits the release of radioactive materials to the environs subsequent to the occurrence of a postulated LOCA so that the offsite doses are below the "reference values" stated in 10 CFR 100. The design employs the drywell/pressure-suppression features of the BWR/Mark II containment concept. The containment consists of a dual barrier: the primary containment, and the secondary containment. The primary containment is a steel-lined reinforced concrete pressure-suppression system of the over-and-under configuration. The secondary containment is the enclosure that encloses the reactor, and its primary containment, and fuel storage areas.

The primary containment is a seismic Class I structure and is designed to withstand the jet forces resulting from a rupture of a reactor coolant system pipe.

The primary containment has provisions for rendering the containment atmosphere non-flammable by reducing and maintaining the oxygen content to less than 4 percent during normal and accident conditions.

The Concrete Containment consists of an 8 ft thick reinforced concrete basemat, a 6 ft-2 in. thick reinforced concrete cylindrical Suppression Chamber wall, and a 6 ft-2 in. thick reinforced concrete conical Drywell wall, which provide containment, structural support, and radiation shielding functions.

The liner plate is 1/4 inch thick steel. It consists of several sections: the cylinder, dome, and floor. These sections are connected by horizontal channels and angles in order to provide a leak tight barrier in the Containment.

Penetrations are located in the Containment so that systems can pass through the pressure boundary while the Containment function is fulfilled. Each penetration consists of a pipe sleeve with an annular ring welded to it, and embedded in the concrete to resist normal operating and accident loads. The pipe sleeve is also welded to the liner plate as a seal to prevent leakage.

The Containment is equipped with a 12 ft.-2 in. diameter equipment hatch in the Drywell wall, a 12 ft.-2 in. diameter equipment hatch/personnel lock in the Drywell wall, two 4-ft. 4-in. diameter access hatches in the Suppression Chamber wall and a 3 foot diameter CRD removal hatch in the Drywell wall.

The LGS Updated Final Safety Analysis Report (UFSAR) Section 6.2.1 describes functional requirements and capabilities of the containment design including the internal design pressure of 55 psig.

4.4 Containment Inspections

As approved in the Reference 4 NRC Safety Evaluation Report, LGS aligned the Inservice Inspection (ISI) and Containment Inservice Inspection Intervals (CISI). As a result, the next LGS, Units 1 and 2 ISI and CISI intervals began on February 1, 2007. Additionally, LGS, Units 1 and 2 comply with the 2001 Edition through the 2003 Addenda for both the ISI and the CISI program.

IWL examinations were performed in the first CISI interval in accordance with the 1992 Edition, 1992 Addenda of the ASME Section XI Code. These exams were performed in accordance with the five (5) year frequency as defined in IWL-2400.

The results of the most recent Unit 1 IWL inspections of concrete revealed no reportable indications. These inspections were completed in 1R10 (2004). The next IWL concrete containment inspections are scheduled to be completed prior to March 2009, in accordance with the requirements of the 2001 Edition, 2003 Addenda, of ASME Section XI, as modified by 10CFR50.55a.

The results of the most recent Unit 2 IWL inspections of concrete revealed no reportable indications. These inspections were completed in 2R08 (2005). The next IWL concrete containment inspections are scheduled to be completed prior to March 2010, in accordance with the requirements of the 2001 Edition, 2003 Addenda, of ASME Section XI, as modified by 10CFR50.55a.

The results of the most recent Unit 1 IWE examinations have been completed per the code requirements. One (1) recordable indication of a pit in the suppression pool steel liner was isolated from further corrosion by performing a qualified coating repair. The remaining wall thickness under this pit was greater than the required design minimum wall thickness. Previously, one other less severe pit was similarly isolated.

The results of the most recent Unit 2 IWE examinations have not identified any recordable indications; however, the suppression pool has not yet been inspected as part of the first interval IWE inspections. This inspection is scheduled to be completed in 2R10 (2009).

There are no IWE augmented inspections required for either Unit 1 or Unit 2.

There are no relief requests being developed for this interval that will impact containment inspections.

NRC Information Notice 92-20 ("Inadequate Local Leak Rate Testing") addresses the inability to obtain valid local leak rate test results on penetrations which are designed with a stainless steel, two-ply bellows. There are no bellows of similar design within the LGS, Units 1 and 2 Appendix J scope.

LGS implements a safety-related coatings program that ensures qualified coating systems are used inside primary containment. The program assures that safety-related coatings are selected, procured, applied and inspected in a manner that conforms to the applicable 10 CFR 50 Appendix B criteria. Unqualified coatings are controlled and tracked to ensure that emergency core cooling systems (ECCS) will not be adversely affected by coating debris following an accident, and to assure coatings will not cause any adverse effects on Systems, Structures, and Components (SSCs) safety functions. The program objective is to conform to licensee commitments made in response to Generic Letter 98-04. Coatings are also monitored in accordance with a formal Maintenance Rule (10 CFR 50.65) condition monitoring program. Engineering reviews and evaluates the results of coating condition examinations performed by examiners qualified in accordance with ASTM D 4537, 1991 Edition.

Based on the above discussion, the ASME Section XI containment inspections and the safety-related coatings program are intended to provide a high degree of assurance that any degradation of the containment structure is identified and corrected before a containment leakage path is introduced.

4.5 Risk Analysis

As discussed in Attachment 4, the Probabilistic Risk Assessment results demonstrate a very small impact in risk associated with the one-time extension of the containment Type A ILRT from 10 to 15 years. The risk assessment follows the guidelines from NEI 94-01 (Reference 1), the methodology used in EPRI TR-104285 (Reference 5), the NEI Interim Guidance for Performing Risk Impact Assessments In Support of One-Time Extensions for Containment Integrated Leakage Rate Test Surveillance Intervals (References 6 and 7), the NRC regulatory guidance on the use of Probabilistic Risk Assessment (PRA) findings and risk insights in support of a request for a plant's licensing basis as outlined in Regulatory Guide 1.174 (Reference 8), and the methodology used for Calvert Cliffs to estimate the likelihood and risk implications of corrosion-induced leakage of steel liners going undetected during the extended test interval. 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 (Reference 9).

The following is a brief summary of some of the key aspects of the Type A ILRT interval extension risk analysis from 10 to 15 years:

- Regulatory Guide 1.174 provides guidance for determining the risk impact of plant-specific changes to the licensing basis. Regulatory Guide 1.174 defines very small changes in risk as resulting in increases of CDF (Core Damage Frequency) below 10-6/yr and increases in LERF (Large Early Release Frequency) below 10-7/yr. Since the ILRT does not impact CDF, the relevant criterion is LERF. The increase in internal events LERF resulting from a change in the Type A ILRT test interval from three in ten years to one in fifteen years is estimated as 4.31E-8/yr using the NEI guidance as written, and at 1.43E-8/yr using the EPRI Expert Elicitation methodology. In either case, the estimated change in LERF is determined to be "very small" using the acceptance guidelines of Regulatory Guide 1.174.

- The change in Type A test frequency to once-per-fifteen-years, measured as an increase to the total integrated plant risk for those accident sequences influenced by Type A testing, is 0.066 person-rem/yr using the NEI guidance, and drops to 0.013 person-rem/yr using the EPRI Expert Elicitation methodology. Therefore, in either case, the risk impact when compared to other severe accident risks is negligible.

- The increase in the Conditional Containment Failure Frequency from the three in ten year interval to one in fifteen year interval is about 1.2% using the NEI guidance, and drops to about 0.4% using the EPRI Expert Elicitation methodology. Although no official acceptance criteria exist for this risk metric, it is judged to be very small.

- Since the increase in LERF falls well below the "small change" category using the acceptance guidelines of Regulatory Guide 1.174, a detailed examination of the external events impact is not required, nor would it change the conclusions from this assessment.

Therefore, increasing the Type A ILRT interval to 15 years is considered to be insignificant since it represents a very small change to the Limerick Generating Station risk profile.

4.6 Primary Containment Pressure Suppression Testing

In the Reference 2 NRC Safety Evaluation Report which approved the use of 10 CFR 50, Appendix J, Option B for LGS, Units 1 and 2, the NRC evaluated the revisions to the scheduling of the drywell-to-suppression chamber bypass leakage test and the drywell-to-suppression chamber vacuum breaker leakage test. The amendment evaluated the proposal to extend the drywell-to-suppression chamber bypass leakage test to a ten-year frequency. This change also included conducting the drywell-to-suppression chamber vacuum breaker leakage tests during those refueling outages when the drywell-to-suppression chamber bypass leakage test is not performed (24-month frequency). This requirement is contained in TS 4.6.2.1.e, which requires that the drywell-to-suppression chamber bypass leak tests be conducted to coincide with the Type A test (ILRT).

This proposed change will extend the drywell-to-suppression chamber bypass leakage test frequency to once in 15 years. A review of the past test history for the drywell-to-suppression chamber bypass leakage test has identified no failures. Therefore, extending this test to a 15 year frequency is acceptable. The following are the test results:

Unit 1 (Acceptance - 0.005 sq. ft.)	Unit 2 (Acceptance - 0.005 sq. ft.)
1984 - 0.00026	1989 - 0.000069
1987 - 0.000051	1993 - 0.000076
1990 - 0.000278	1999 - 0.000012
1998 - 0.000075	

No frequency change is required for the drywell-to-suppression chamber vacuum breaker leakage tests, because these tests are conducted independently of the Type A ILRT.

Additionally, the proposed changes do not modify the acceptance criteria of either of these tests.

5.0 REGULATORY ANALYSIS

5.1 NO SIGNIFICANT HAZARDS CONSIDERATION

Exelon Generation Company, LLC (EGC) has evaluated the proposed change to the Technical Specifications (TS) for Limerick Generating Station (LGS), Units 1 and 2 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 6.8.4.g ("Primary Containment Leakage Rate Testing Program") of the LGS, Units 1 and 2 TS to reflect a one-time extension to the 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 10 to 15 years, which is "no later than May 15, 2013" for LGS, Unit 1 and is "no later than May 21, 2014" for Unit 2.

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 LGS, Units 1 and 2 containments will not exceed allowable leakage rate values specified in the TS and will continue to perform their design function following an accident. The risk assessment of the proposed change has concluded that there is an insignificant increase in Large Early Release Frequency, Person-Rem, and Conditional Containment Failure Frequency. Additionally, containment inspections have also been performed which demonstrate the continued structural integrity of the primary containment and will be performed in the future as required by the ASME Code.

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 ILRTs for LGS, Units 1 and 2 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.

EGC has conducted a risk assessment to determine the impact of a change to the LGS, Units 1 and 2 Type A ILRT from 10 to 15 years. This risk assessment measured the impact to the Large Early Release Frequency, Person-Rem, and Conditional Containment Failure Frequency. This assessment indicated that the proposed LGS, Units 1 and 2 Type A ILRT interval extension has a very 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, EGC 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 6.8.4.g to reflect a one-time extension to the LGS, Units 1 and 2 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 LGS, Units 1 and 2 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 LGS, Units 1 and 2 TS a change that is similar to changes (i.e., extend Type A ILRT from 10 to 15 years) approved by the NRC for Peach Bottom Atomic Power Station, Unit 3 (Reference 10), Three Mile Island, Unit 1 (Reference 11), Vermont Yankee Nuclear Power Station (Reference 12), and Cooper Nuclear Station (Reference 13).

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) U. S. Nuclear Regulatory Commission letter dated January 24, 1997, "Limerick Generating Station, Units 1 and 2 (TAC NOS. M96117 and M96118)"

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

(4) U. S. Nuclear Regulatory Commission letter dated January 24, 2007, "Limerick Generation Station, Units 1 and 2 – Relief Requests 13R-01 For Alignment of Inservice Inspection and Containment Inservice Inspection (TAC NOS. MD2727 AND MD 2728)"

(5) Electric Power Research Institute, "Risk Impact Assessment of Revised Containment Leak Rate Testing Intervals", EPRI TR-104285, August 1994.

(6) Letter from A. Pietrangelo (NEI) to NEI Administrative Points of Contact, "Interim Guidance for Performing Risk Impact Assessments in Support of One-Time Extensions for Containment Integrated Leak Rate Test Surveillance Intervals", November 13, 2001.

(7) Letter from A. Pietrangelo (NEI) to NEI Administrative Points of Contact, "One-Time Extension of Containment Integrated Leak Rate Test Interval – Additional Information", November 30, 2001.

(8) Regulatory Guide 1.174, "An Approach for Using Probabilistic Risk Assessment in Risk-Informed Decisions on Plant-Specific Changes to the Licensing Basis", July 1998.

(9) Electric Power Research Institute, "Risk Impact Assessment of Extended Integrated Leak Rate Testing Intervals", 1009325, Revision 1, December 2005.

(10) U. S. Nuclear Regulatory Commission letter dated October 4, 2001, "Peach Bottom Atomic Power Station, Unit 3 - Issuance of Amendment RE: Extension of the Containment Integrated Leak Rate Test (TAC NO. MB2094)"

(11) U. S. Nuclear Regulatory Commission letter dated August 13, 2003, "Three Mile Island Nuclear Station, Unit 1 (TMI-1), RE: Deferral of Containment Integrated Leakage Rate Test (TAC NO. MB6487)"

(12) U. S. Nuclear Regulatory Commission letter dated August 31, 2005, "Vermont Yankee Nuclear Power Station – Issuance of Amendment RE: One-Time Extension of Integrated Leak Rate Test Interval (TAC NO. MC4662)"

(13) U. S. Nuclear Regulatory Commission letter dated October 3, 2006, "Cooper Nuclear Station – Issuance of Amendment RE: Additional Extension of Appendix J, Type A, Integrated Leak Rate Test Interval (TAC NO. MC9732)"

ATTACHMENT 2

MARKUP OF PROPOSED TECHNICAL SPECIFICATION PAGE CHANGE

Revised TS Pages

6-14c (Units 1 and 2)

g. Primary Containment Leakage Rate Testing Program

A program shall be established to implement the leakage rate testing of the containment as required by 10 CFR 50.54 (o) and 10 CFR 50, Appendix J, Option B, as modified by approved exemptions. This program shall be in accordance with the guidelines contained in Regulatory Guide 1.163 "Performance-~~Based~~ Containment Leakage Test program," dated September 1995. INSERT A

The peak calculated containment internal pressure for the design basis loss of coolant accident, P_a, is 44.0 psig.

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

Leakage rate acceptance criteria are:

a. Primary Containment leakage rate acceptance criterion is less than or equal to 1.0 L_a. During the first unit startup following testing in accordance with this program, the leakage rate acceptance criteria are less than or equal to 0.60 L_a for the Type B and Type C tests and less than or equal to 0.75 L_a for Type A tests;

b. Air lock testing acceptance criteria are:

1) Overall airlock leakage rate is less than or equal to 0.05 L_a when tested at greater than or equal to P_a.

2) Seal leakage rate is less than or equal to 5 scf per hour when the gap between the door seals is pressurized to 10 psig.

The provisions of Specification 4.0.2 do not apply to the test frequencies specified in the Primary Containment Leakage Rate Testing Program.

The provisions of Specification 4.0.3 are applicable to the tests described in the Primary Containment Leakage Rate Testing Program.

h. Technical Specifications (TS) Bases Control Program

This program provides a means for processing changes to the Bases of these Technical Specifications.

a. Changes to the Bases of the TS shall be made under appropriate administrative controls and reviews.

b. Licensees may make changes to Bases without prior NRC approval provided the changes do not require either of the following:

A change in the TS incorporated in the license; or

A change to the UFSAR or Bases that requires NRC approval pursuant to 10 CFR 50.59.

c. The Bases Control Program shall contain provisions to ensure that the Bases are maintained consistent with the UFSAR.

d. Proposed changes that meet the criteria of b. above shall be reviewed and approved by the NRC prior to implementation. Changes to the Bases implemented without prior NRC approval shall be provided to the NRC on a frequency consistent with 10 CFR 50.71(e).

INSERT A

, as modified by the following exception to NEI 94-01, Rev. 0, "Industry Guideline for Implementing Performance-Based Option of 10 CFR 50, Appendix J":

a. Section 9.2.3: The first Type A test performed after May 15, 1998 shall be performed no later than May 15, 2013.

g. Primary Containment Leakage Rate Testing Program

A program shall be established to implement the leakage rate testing of the containment as required by 10 CFR 50.54 (o) and 10 CFR 50, Appendix J, Option B, as modified by approved exemptions. This program shall be in accordance with the guidelines contained in Regulatory Guide 1.163 "Performance-Based Containment Leakage Test program," dated September 1995 or INSERT B

The peak calculated containment internal pressure for the design basis loss of coolant accident, P_a, is 44.0 psig.

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

Leakage rate acceptance criteria are:

a. Primary Containment leakage rate acceptance criterion is less than or equal to 1.0 L_a. During the first unit startup following testing in accordance with this program, the leakage rate acceptance criteria are less than or equal to 0.60 L_a for the Type B and Type C tests and less than or equal to 0.75 L_a for Type A tests;

b. Air lock testing acceptance criteria are:

1) Overall airlock leakage rate is less than or equal to 0.05 L_a when tested at greater than or equal to P_a.

2) Seal leakage rate is less than or equal to 5 scf per hour when the gap between the door seals is pressurized to 10 psig.

The provisions of Specification 4.0.2 do not apply to the test frequencies specified in the Primary Containment Leakage Rate Testing Program.

The provisions of Specification 4.0.3 are applicable to the tests described in the Primary Containment Leakage Rate Testing Program.

h. Technical Specifications (TS) Bases Control Program

This program provides a means for processing changes to the Bases of these Technical Specifications.

a. Changes to the Bases of the TS shall be made under appropriate administrative controls and reviews.

b. Licensees may make changes to Bases without prior NRC approval provided the changes do not require either of the following:

A change in the TS incorporated in the license; or

A change to the UFSAR or Bases that requires NRC approval pursuant to 10 CFR 50.59.

c. The Bases Control Program shall contain provisions to ensure that the Bases are maintained consistent with the UFSAR.

d. Proposed changes that meet the criteria of b. above shall be reviewed and approved by the NRC prior to implementation. Changes to the Bases implemented without prior NRC approval shall be provided to the NRC on a frequency consistent with 10 CFR 50.71(e).

INSERT B

, as modified by the following exception to NEI 94-01, Rev. 0, "Industry Guideline for Implementing Performance-Based Option of 10 CFR 50, Appendix J":

a. Section 9.2.3: The first Type A test performed after May 21, 1999 shall be performed no later than May 21, 2014.

ATTACHMENT 3

RETYPED PAGE FOR TECHNICAL SPECIFICATION CHANGE

<u>Retyped TS Pages</u>

6-14c (Units 1 and 2)

g. Primary Containment Leakage Rate Testing Program

A program shall be established to implement the leakage rate testing of the containment as required by 10 CFR 50.54 (o) and 10 CFR 50, Appendix J, Option B, as modified by approved exemptions. This program shall be in accordance with the guidelines contained in Regulatory Guide 1.163 "Performance-Based Containment Leakage 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 50, Appendix J”:

a. Section 9.2.3: The first Type A test performed after May 15, 1998 shall be performed no later than May 15, 2013.

The peak calculated containment internal pressure for the design basis loss of coolant accident, P_a, is 44.0 psig.

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

Leakage rate acceptance criteria are:

a. Primary Containment leakage rate acceptance criterion is less than or equal to 1.0 L_a. During the first unit startup following testing in accordance with this program, the leakage rate acceptance criteria are less than or equal to 0.60 L_a for the Type B and Type C tests and less than or equal to 0.75 L_a for Type A tests;

b. Air lock testing acceptance criteria are:

1) Overall airlock leakage rate is less than or equal to 0.05 L_a when tested at greater than or equal to P_a.

2) Seal leakage rate is less than or equal to 5 scf per hour when the gap between the door seals is pressurized to 10 psig.

The provisions of Specification 4.0.2 do not apply to the test frequencies specified in the Primary Containment Leakage Rate Testing Program.

The provisions of Specification 4.0.3 are applicable to the tests described in the Primary Containment Leakage Rate Testing Program.

h. Technical Specifications (TS) Bases Control Program

This program provides a means for processing changes to the Bases of these Technical Specifications.

a. Changes to the Bases of the TS shall be made under appropriate administrative controls and reviews.

b. Licensees may make changes to Bases without prior NRC approval provided the changes do not require either of the following:

A change in the TS incorporated in the license; or

A change to the UFSAR or Bases that requires NRC approval pursuant to 10 CFR 50.59.

c. The Bases Control Program shall contain provisions to ensure that the Bases are maintained consistent with the UFSAR.

d. Proposed changes that meet the criteria of b. above shall be reviewed and approved by the NRC prior to implementation. Changes to the Bases implemented without prior NRC approval shall be provided to the NRC on a frequency consistent with 10 CFR 50.71(e).

PROCEDURES AND PROGRAMS (Continued)

g. Primary Containment Leakage Rate Testing Program

A program shall be established to implement the leakage rate testing of the containment as required by 10 CFR 50.54 (o) and 10 CFR 50, Appendix J, Option B, as modified by approved exemptions. This program shall be in accordance with the guidelines contained in Regulatory Guide 1.163 "Performance-Based Containment Leakage 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 50, Appendix J":

a. Section 9.2.3: The first Type A test performed after May 21, 1999 shall be performed no later than May 21, 2014.

The peak calculated containment internal pressure for the design basis loss of coolant accident, P_a, is 44.0 psig.

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

Leakage rate acceptance criteria are:

a. Primary Containment leakage rate acceptance criterion is less than or equal to 1.0 L_a. During the first unit startup following testing in accordance with this program, the leakage rate acceptance criteria are less than or equal to 0.60 L_a for the Type B and Type C tests and less than or equal to 0.75 L_a for Type A tests;

b. Air lock testing acceptance criteria are:

1) Overall airlock leakage rate is less than or equal to 0.05 L_a when tested at greater than or equal to P_a.

2) Seal leakage rate is less than or equal to 5 scf per hour when the gap between the door seals is pressurized to 10 psig.

The provisions of Specification 4.0.2 do not apply to the test frequencies specified in the Primary Containment Leakage Rate Testing Program.

The provisions of Specification 4.0.3 are applicable to the tests described in the Primary Containment Leakage Rate Testing Program.

h. Technical Specifications (TS) Bases Control Program

This program provides a means for processing changes to the Bases of these Technical Specifications.

a. Changes to the Bases of the TS shall be made under appropriate administrative controls and reviews.

b. Licensees may make changes to Bases without prior NRC approval provided the changes do not require either of the following:

A change in the TS incorporated in the license; or

A change to the UFSAR or Bases that requires NRC approval pursuant to 10 CFR 50.59.

c. The Bases Control Program shall contain provisions to ensure that the Bases are maintained consistent with the UFSAR.

d. Proposed changes that meet the criteria of b. above shall be reviewed and approved by the NRC prior to implementation. Changes to the Bases implemented without prior NRC approval shall be provided to the NRC on a frequency consistent with 10 CFR 50.71(e).

ATTACHMENT 4

Risk Assessment for LGS, Units 1 and 2 to Support ILRT (Type A) Interval Extension Request

RM DOCUMENTATION NO.	LG-2006-LAR-01	REV: 0	PAGE NO. 1

STATION: LIMERICK

UNIT(S) AFFECTED: 1 and 2

TITLE: Risk Assessment for Limerick Unit 1 and Unit 2 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 Limerick Unit 1 and Limerick Unit 2 containment Type A integrated leak rate test (ILRT) interval from 10 years to 15 years.

Internal RM Documentation

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

Prepared by: Donald E. Vanover / Donald E. Vanover / 12/14/06
Print / Sign / Date

Reviewed by: Robert J. Wolfgang / Robert J. Wolfgang / 12/14/06
Print / Sign / Date

Method of Review: [X] Detailed [ ] Alternate

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

Approved by: Gregory A. Krueger / [signature] / 12/18/06
Print / Sign / Date

External RM Documentation

Reviewed by: ____ / ____ / ____
Print / Sign / Date

Approved by: 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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Section **Page**

1.0 PURPOSE OF ANALYSIS

1.1 Purpose

The purpose of this analysis is to provide an assessment of the risk associated with implementing a one-time extension of the Limerick Generating Station Units 1 and 2 containment Type A integrated leak rate test (ILRT) interval from ten years to fifteen years. The extension would allow for substantial cost savings as the ILRT could be deferred for additional scheduled refueling outages. The risk assessment follows the guidelines from NEI 94-01 [1], the methodology used in EPRI TR-104285 [2], the NEI Interim Guidance for Performing Risk Impact Assessments In Support of One-Time Extensions for Containment Integrated Leakage Rate Test Surveillance Intervals [3, 21], the NRC regulatory guidance on the use of Probabilistic Risk Assessment (PRA) findings and risk insights in support of a request for a plant's licensing basis as outlined in Regulatory Guide (RG) 1.174 [4], and the methodology used for Calvert Cliffs to estimate the likelihood and risk implications of corrosion-induced leakage of steel liners going undetected during the extended test interval [19]. 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 [22].

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 $1.0L_a$ (allowable leakage).

The basis for the current 10-year test interval is provided in Section 11.0 of NEI 94-01, Revision 0, and was established in 1995 during development of the performance-based

Option B to Appendix J. Section 11.0 of NEI 94-01 states that NUREG-1493 [5], "Performance-Based Containment Leak Test Program," September 1995, provides the technical basis to support rulemaking to revise leakage rate testing requirements contained in Option B to Appendix J. The basis consisted of qualitative and quantitative assessments of the risk impact (in terms of increased public dose) associated with a range of extended leakage rate test intervals. To supplement the NRC's rulemaking basis, NEI undertook a similar study. The results of that study are documented in Electric Power Research Institute (EPRI) Research Project Report TR-104285.

The NRC report, Performance Based Leak Test Program, NUREG-1493 [5], analyzed the effects of containment leakage on the health and safety of the public and the benefits realized from the containment leak rate testing. In that analysis, it was determined for a comparable BWR plant, that increasing the containment leak rate from the nominal 0.5 percent per day to 5 percent per day leads to a barely perceptible increase in total population exposure, and increasing the leak rate to 50 percent per day increases the total population exposure by less than 1 percent. Consequently, extending the ILRT interval should not lead to any substantial increase in risk. The current analysis is being performed to confirm these conclusions based on Limerick 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 ILRT interval extension risk assessment for Limerick 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

The acceptance guidelines in RG 1.174 are used to assess the acceptability of this one-time extension of the Type A test interval beyond that established during the Option B rulemaking of Appendix J. RG 1.174 defines very small changes in the risk-acceptance guidelines as increases in core damage frequency (CDF) less than 10^{-6} per reactor year and increases in large early release frequency (LERF) less than 10^{-7} per reactor year. Since the Type A test does not impact CDF for Limerick, the relevant criterion is the change in LERF. RG 1.174 also defines small changes in LERF as below 10^{-6} per reactor year provided that the total from all contributors (including external events) can be reasonably shown to be less than 10^{-5} per reactor year. RG 1.174 discusses defense-in-depth and encourages the use of risk analysis techniques to help ensure and show that key principles, such as the defense-in-depth philosophy, are met. Therefore, the increase in the conditional containment failure probability (CCFP) that helps to ensure that the defense-in-depth philosophy is maintained is also calculated.

In addition, the total annual risk (person rem/yr population dose) is examined to demonstrate the relative change in this parameter based on the precedent set by previous submittals for ILRT extensions [6, 20, 23]. (No criteria have been established for this parameter change.)

2.0 METHODOLOGY

A simplified bounding analysis approach consistent with the EPRI approach is used for evaluating the change in risk associated with increasing the test interval to fifteen years [22]. The approach is consistent with that presented in NEI Interim Guidance [3, 21], EPRI TR-104285 [2], NUREG-1493 [5] and the Calvert Cliffs liner corrosion analysis [19]. The analysis uses results from a Level 2 analysis of core damage scenarios from the current Limerick PRA model and subsequent containment response resulting in various fission product release categories (including no or negligible release). The Limerick Unit 1 model is used in this analysis, but since there is no significant difference between the two Unit models, this risk assessment is applicable to Limerick Units 1 and 2. Confidence in the validity of this assumption can be obtained by examining the comparison of the Unit 1 and Unit 2 Level 1 model results shown in Table 2-1 below, and the Level 2 model results shown in Table 2-2 below.

The six general steps of this assessment are as follows:

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

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

- Consistent with the other industry containment leak risk assessments, the Limerick assessment uses population dose as one of the risk measures. The other risk measures used in the Limerick assessment are LERF and the conditional containment failure probability (CCFP) to demonstrate that the acceptance guidelines from RG 1.174 are met.
- This evaluation for Limerick uses ground rules and methods to calculate changes in risk metrics that are similar to those used in the EPRI approach.

Table 2-1

SUMMARY OF THE LIMERICK CORE DAMAGE FREQUENCY BY ACCIDENT SEQUENCE SUBCLASS

Accident Class Designator	Subclass	Definition	LGS Unit 1 Frequency (per year)	LGS Unit 2 Frequency (per year)
Class I	A	Accident sequences involving loss of inventory makeup in which the reactor pressure remains high.	1.19E-06	1.19E-06
	B	Accident sequences involving a loss of offsite power and loss of coolant inventory makeup.	1.17E-06	1.17E-06
	C	Accident sequences involving a loss of coolant inventory induced by an ATWS sequence with containment intact.	2.56E-08	2.56E-08
	D	Accident sequences involving a loss of coolant inventory makeup in which reactor pressure has been successfully reduced to 200 psi.	9.68E-08	9.68E-08
	E	Accident sequences involving loss of inventory makeup in which the reactor pressure remains high and DC power is unavailable.	3.39E-09	3.39E-09
Class II	A	Accident sequences involving a loss of containment heat removal with the RPV initially intact; core damage; core damage induced post containment failure.	1.02E-07	1.01E-07
	F	Class IIA and IIL except that the vent operates as designed; loss of makeup occurs at some time following vent initiation. Suppression pool saturated but intact.	6.78E-07	6.78E-07
	L	Accident sequences involving a loss of containment heat removal with the RPV breached but no initial core damage; core damage induced post containment failure. (Note that this is a new category for the 2004 update, and is grouped with Class IIA for transfer to the Level 2 model for consistency with previous treatment in the Level 2 model evaluation.)	3.61E-08	3.61E-08

Table 2-1

SUMMARY OF THE LIMERICK CORE DAMAGE FREQUENCY BY ACCIDENT SEQUENCE SUBCLASS

Accident Class Designator	Subclass	Definition	LGS Unit 1 Frequency (per year)	LGS Unit 2 Frequency (per year)
Class III (LOCA)	A	Accident sequences leading to core damage conditions initiated by vessel rupture where the containment integrity is not breached in the initial time phase of the accident.	9.50E-09	9.50E-09
	B	Accident sequences initiated or resulting in small or medium LOCAs for which the reactor cannot be depressurized prior to core damage occurring.	3.08E-08	3.08E-08
	C	Accident sequences initiated or resulting in medium or large LOCAs for which the reactor is a low pressure and no effective injection is available.	4.55E-08	4.55E-08
	D	Accident sequences which are initiated by a LOCA or RPV failure and for which the vapor suppression system is inadequate, challenging the containment integrity with subsequent failure of makeup systems.	1.99E-08	1.99E-08
Class IV (ATWS)	A	Accident sequences involving failure of adequate shutdown reactivity with the RPV initially intact; core damage induced post containment failure.	2.22E-07	2.22E-07
	L	Accident sequences involving failure of adequate shutdown reactivity with the RPV initially breached; core damage induced post containment failure. (Note that this is a new category for the 2004 update, and is grouped with Class IIA for transfer to the Level 2 model for consistency with previous treatment in the Level 2 model evaluation.)	4.29E-08	4.29E-08
Class V	---	Unisolated LOCA outside containment.	3.24E-08	3.24E-08

Table 2-2

SUMMARY OF THE LIMERICK LEVEL 2 MODEL RESULTS BY RELEASE CATEGORY

Release Category [(1)]	LGS Unit 1 Frequency (per year)	LGS Unit 2 Frequency (per year)
M/I	1.39E-06	1.39E-06
L/L	2.28E-07	2.28E-07
M/L	1.98E-07	1.98E-07
LLE	1.46E-07	1.46E-07
L/I	1.09E-07	1.09E-07
M/E	9.27E-08	9.27E-08
LLL	8.92E-08	8.92E-08
H/E	6.80E-08	6.80E-08
L/E	3.09E-08	3.09E-08
H/L	1.18E-08	1.18E-08
H/I	8.53E-09	8.53E-09
LLI	1.42E-09	1.43E-09
Total[(2)]:	2.37E-06	2.37E-06

(1) Refer to Table 2-3 for the release category classification scheme used in the Limerick Level 2 analysis.

(2) The difference between this value and the total CDF value of 3.70E-06 is assigned to the Containment Intact (OK) category.

Table 2-3

RELEASE SEVERITY AND TIMING CLASSIFICATION SCHEME (SEVERITY, TIMING)

Release Severity Term Release Fraction		Release Timing	
Classification Category	Cs Iodide % in Release	Classification Category	Time of Release(1)
High (H)	Greater than 10	Late (L)	Greater than 24 hours
Moderate (M)	1 to 10	Intermediate (I)	6 to 24 hours
Low (L)	0.1 to 1	Early (E)	Less than 6 hours
Low-low (LL)	Less than 0.1		
No Iodine (OK)	0		

(1) Relative to the declaration of a General Emergency.

3.0 GROUND RULES

The following ground rules are used in the analysis:

- The Limerick Level 1 and Level 2 internal events PRA models provide representative results.
- It is appropriate to use the Limerick internal events PRA model as a gauge to effectively describe the risk change attributable to the ILRT extension. It is reasonable to assume that the impact from the ILRT extension (with respect to percent increases in population dose) will not substantially differ if fire and seismic events were to be included in the calculations.
- Dose results for the containment failures modeled in the PRA can be characterized by information provided in NUREG/CR-4551 [9]. They are estimated by scaling the NUREG/CR-4551 results by population differences for Limerick compared to the NUREG/CR-4551 reference plant.
- Accident classes describing radionuclide release end states are defined consistent with EPRI methodology [2] and are summarized in Section 4.2.
- The representative containment leakage for Class 1 sequences is $1L_a$. Class 3 accounts for increased leakage due to Type A inspection failures.
- The representative containment leakage for Class 3a sequences is $10L_a$, based on the previously approved methodology performed for Indian Point Unit 3 [6, 7].
- The representative containment leakage for Class 3b sequences is $35L_a$, based on the previously approved methodology [6, 7].
- The Class 3b can be very conservatively categorized as LERF based on the previously approved methodology [6, 7]. The Class 3b category increase is used as a surrogate for LERF in this application even though the releases associated with a $35L_a$ release would not necessarily be consistent with a "Large" release for Limerick.
- The impact on population doses from containment bypass scenarios is not altered by the proposed ILRT extension, but is accounted for in the EPRI methodology as

a separate entry for comparison purposes. Since the containment bypass contribution to population dose is fixed, no changes on the conclusions from this analysis will result from this separate categorization.

- The reduction in ILRT frequency does not impact the reliability of containment isolation valves to close in response to a containment isolation signal.
- The use of estimated 2010 population data is adequate for this analysis. Scaling the year 2010 population data to the date of the next ILRT test if extended beyond the current due date would not significantly impact the quantitative results, nor would it change the conclusions.
- An evaluation of the risk impact of the ILRT on shutdown risk is addressed using the generic results from EPRI TR-105189 [8].

4.0 INPUTS

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

4.1 General Resources Available

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

1. NUREG/CR-3539 [10]
2. NUREG/CR-4220 [11]
3. NUREG-1273 [12]
4. NUREG/CR-4330 [13]
5. EPRI TR-105189 [8]
6. NUREG-1493 [5]
7. EPRI TR-104285 [2]
8. NUREG-1150 [14] and NUREG/CR-4551 [9]
9. NEI Interim Guidance [3, 21]
10. Calvert Cliffs liner corrosion analysis [19]
11. EPRI 1009325 [22]

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. The eighth study provides an ex-plant consequence analysis for a 50-mile radius surrounding a plant that is used as the bases for the consequence analysis of the ILRT interval extension for Limerick. The ninth study includes the NEI recommended methodology for evaluating the risk associated with obtaining a one-time extension of the ILRT interval. The tenth study addresses the impact of age-related degradation of the containment liners on ILRT evaluations. Finally, the last study complements the previous EPRI report [2], integrates the NEI interim guidance, and provides the results of an expert elicitation process to determine the relationship between pre-existing containment leakage probability and magnitude.

<u>NUREG/CR-3539 [10]</u>

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.

<u>NUREG/CR-4220 [11]</u>

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

This calculation presented in NUREG/CR-4220 is called an "upper bound" estimate for BWRs (presumably meaning "inerted" BWR containment designs).

<u>NUREG-1273 [12]</u>

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.

<u>NUREG/CR-4330 [13]</u>

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."

<u>EPRI TR-105189 [8]</u>

The EPRI study TR-105189 is useful to the ILRT test interval extension risk assessment because this EPRI study provides insight regarding the impact of containment testing on shutdown risk. This study performed a quantitative evaluation (using the EPRI ORAM software) for two reference plants (a BWR-4 and a PWR) of the impact of extending ILRT and LLRT test intervals on shutdown risk.

The result of the study concluded that a small but measurable safety benefit is realized from extending the test intervals. For the BWR, the benefit from extending the ILRT frequency from 3 per 10 years to 1 per 10 years was calculated to be a reduction of approximately 1E-7/yr in the shutdown core damage frequency. This risk reduction is due to the following issues:

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

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

<u>NUREG-1493 [5]</u>

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.

<u>EPRI TR-104285 [2]</u>

Extending the risk assessment impact beyond shutdown (the earlier EPRI TR-105189 study), the EPRI TR-104285 study is a quantitative evaluation of the impact of extending 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 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>Release Category Definitions</u>

Table 4.1-1 defines the accident classes used in the ILRT extension evaluation, which is consistent with the EPRI/NEI methodology [2]. These containment failure classifications are used in this analysis to determine the risk impact of extending the Containment Type A test interval as described in Section 5 of this report.

Table 4.1-1

EPRI/NEI CONTAINMENT FAILURE CLASSIFICATIONS [2]

Class	Description
1	Containment remains intact including accident sequences that do not lead to containment failure in the long term. The release of fission products (and attendant consequences) is determined by the maximum allowable leakage rate values L_a, under Appendix J for that plant
2	Containment isolation failures (as reported in the IPEs) include those accidents in which there is a failure to isolate the containment.
3	Independent (or random) isolation failures include those accidents in which the pre-existing isolation failure to seal (i.e., provide a leak-tight containment) is not dependent on the sequence in progress.
4	Independent (or random) isolation failures include those accidents in which the pre-existing isolation failure to seal is not dependent on the sequence in progress. This class is similar to Class 3 isolation failures, but is applicable to sequences involving Type B tests and their potential failures. These are the Type B-tested components that have isolated but exhibit excessive leakage.
5	Independent (or random) isolation failures include those accidents in which the pre-existing isolation failure to seal is not dependent on the sequence in progress. This class is similar to Class 4 isolation failures, but is applicable to sequences involving Type C tests and their potential failures.
6	Containment isolation failures include those leak paths covered in the plant test and maintenance requirements or verified per in service inspection and testing (ISI/IST) program.
7	Accidents involving containment failure induced by severe accident phenomena. Changes in Appendix J testing requirements do not impact these accidents.
8	Accidents in which the containment is bypassed (either as an initial condition or induced by phenomena) are included in Class 8. Changes in Appendix J testing requirements do not impact these accidents.

<u>NUREG-1150 [14] and NUREG/CR 4551 [9]</u>

NUREG-1150 and the technical basis, NUREG/CR-4551, provide an ex-plant consequence analysis for a spectrum of accidents including a severe accident with the containment remaining intact (i.e., Technical Specification leakage). This ex-plant consequence analysis is calculated for the 50-mile radial area surrounding Peach Bottom. The ex-plant calculation can be delineated to total person-rem for each identified Accident Progression Bin (APB) from NUREG/CR-4551. With the Limerick

Level 2 model end-states assigned to one of the NUREG/CR-4551 APBs, it is considered adequate to represent Limerick if the population is scaled to represent Limerick. (The meteorology and site differences other than population are assumed not to play a significant role in this evaluation.)

NEI Interim Guidance [3, 21]

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 dose calculations to support the population dose determination.

The guidance provided in this document builds on the EPRI risk impact assessment methodology [2] and the NRC performance-based containment leakage test program [5], and considers approaches utilized in various submittals, including Indian Point 3 [6,7] (and associated NRC SER) and Crystal River [20].

The approach included in this guidance document is used in the Limerick assessment to determine the estimated increase in risk associated with the ILRT extension. This document includes the bases for the values assigned in determining the probability of leakage for the EPRI Class 3a and 3b scenarios in this analysis as described in Section 5.

Calvert Cliffs Response to Request for Additional Information Concerning the License Amendment for a One-Time Integrated Leakage Rate Test Extension [19].

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

EPRI Risk Impact Assessment of Extended Integrated Leak Rate Testing Intervals [22]

This report presents a risk impact assessment for extending integrated leak rate test (ILRT) surveillance intervals to 15 years and is consistent in nature with the NEI interim guidance. This risk impact assessment complements the previous EPRI report, TR-104285, *Risk Impact Assessment of Revised Containment Leak Rate Testing Intervals*. The earlier report considered changes to local leak rate testing intervals as well as changes to ILRT testing intervals. The original risk impact assessment considers the change in risk based on population dose, whereas the revision considers dose as well as large early release frequency (LERF) and conditional containment failure probability (CCFP). This report deals with changes to ILRT testing intervals and is intended to provide bases for supporting changes to industry (NEI) and regulatory (NRC) guidance on ILRT surveillance intervals.

The risk impact assessment using the Jeffery's Non-Informative Prior statistical method is further supplemented with a sensitivity case using expert elicitation performed to address conservatisms. The expert elicitation is used to determine the relationship between pre-existing containment leakage probability and magnitude. The results of the expert elicitation process from this report are used as a separate sensitivity investigation for the Limerick analysis presented here in Section 6.2.

4.2 Plant-Specific Inputs

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

- Level 1 Model results [16]
- Level 2 Model results [16]
- Population within a 50-mile radius [17]
- ILRT results to demonstrate adequacy of the administrative and hardware issues [(1)]

Limerick Internal Events Level 1 PRA Model

The current Level 1 PRA model is an event tree / linked fault tree model characteristic of the as-built, as-operated plant. The total internal events core damage frequency (CDF) used in this analysis is 3.70E-06/yr for both Unit 1 and Unit 2 [16].

Limerick Internal Events Level 2 PRA Model

The Level 2 Model that is used for Limerick was developed to calculate the LERF contribution as well as the other release categories evaluated in the model. Table 4.2-1 summarizes the pertinent Limerick results in terms of release category. The total Large Early Release Frequency (LERF) which corresponds to the H/E release category in Table 4.2-1 was found to be 6.80E-8/yr. The total release frequency is 2.37E-06/yr. With a total CDF of 3.70E-06/yr, this corresponds to an "OK" release limited to normal leakage of 1.33E-6/yr [16].

(1) Limerick ST-1-060-490-1/2 identifies that the 95% Upper Confidence Level (UCL) leak rate shall be less than 0.75La (0.375% wt/day). The following acceptable leak rates were identified [18]:

	Unit 1 (%wt/day)		Unit 2 (%wt/day)
1984	0.1642	1989	0.2310
1987	0.1468	1993	0.1836
1990	0.2870	1999	0.3272
1998	0.3070		

Since the two most recent Type A tests at Limerick Unit 1 and Unit 2 have been successful, the current Type A test interval requirement is 10 years.

Table 4.2-1

Limerick Level 2 PRA Model Release Categories and Frequencies

Category	Frequency/yr
H/E – High Early (LERF)	6.80E-08
M/E – Medium Early	9.27E-08
L/E – Low Early	3.09E-08
LL/E – Low Low Early	1.46E-07
H/I – High Intermediate	8.53E-09
M/I – Medium Intermediate	1.39E-06
L/I – Low Intermediate	1.09E-07
LL/I – Low Low Intermediate	1.42E-09
H/L – High Late	1.18E-08
M/L – Medium Late	1.98E-07
L/L – Low Late	2.28E-07
LL/L – Low Low Late	8.92E-08
Total Release Frequency	**2.37E-06**
Core Damage Frequency	**3.70E-06**

Population Dose Calculations

The population dose is calculated by using data provided in NUREG/CR-4551 and adjusting the results for Limerick. Each accident sequence was associated with an applicable collapsed Accident Progression Bin (APB) from NUREG/CR-4551. The collapsed APBs are characterized by 5 attributes related to the accident progression. Unique combinations of the 5 attributes result in a set of 10 bins that are relevant to the analysis. Information from the Limerick PRA Containment Event Trees (CETs) was used to classify each of the Level 2 sequences using these attributes. The definitions of the 10 collapsed APBs are provided in NUREG/CR-4551 and are reproduced in Table 4.2-2 for references purposes. Table 4.2-3 summarizes the calculated population dose associated with each APB from NUREG/CR-4551.

Table 4.2-2
Collapsed Accident Progression Bin (APB) Descriptions [9]

Collapsed APB Number	Description
1	CD, VB, Early CF, WW Failure, RPV Pressure > 200 psi at VB Core damage occurs followed by vessel breach. The containment fails early in the wetwell (i.e., either before core damage, during core damage, or at vessel breach) and the RPV pressure is greater than 200 psi at the time of vessel breach (this means Direct Containment Heating (DCH) is possible).
2	CD, VB, Early CF, WW Failure, RPV Pressure < 200 psi at VB Core Damage occurs followed by vessel breach. The containment fails early in the wetwell (i.e., either before core damage, during core damage, or at vessel breach) and the RPV pressure is less than 200 psi at the time of vessel breach (this means DCH is not possible).
3	CD, VB, Early CF, DW Failure, RPV Pressure > 200 psi at VB Core damage occurs followed by vessel breach. The containment fails early in the drywell (i.e., either before core damage, during core damage, or at vessel breach) and the RPV pressure is greater than 200 psi at the time of vessel breach (this means DCH is possible).
4	CD, VB, Early CF, DW Failure, RPV Pressure < 200 psi at VB Core Damage occurs followed by vessel breach. The containment fails early in the drywell (i.e., either before core damage, during core damage, or at vessel breach) and the RPV pressure is less than 200 psi at the time of vessel breach (this means DCH is not possible).
5	CD, VB, Late CF, WW Failure, N/A Core Damage occurs followed by vessel breach. The containment fails late in the wetwell (i.e., after vessel breach during Molten Core-Concrete Interaction (MCCI)) and the RPV pressure is not important since, even if DCH occurred, it did not fail containment at the time it occurred.
6	CD, VB, Late CF, DW Failure, N/A Core Damage occurs followed by vessel breach. The containment fails late in the drywell (i.e., after vessel breach during MCCI) and the RPV pressure is not important since, even if DCH occurred, it did not fail containment at the time it occurred.

Table 4.2-2
Collapsed Accident Progression Bin (APB) Descriptions [9]

Collapsed APB Number	Description
7	CD, VB, No CF, Vent, N/A Core Damage occurs followed by vessel breach. The containment never structurally fails, but is vented sometime during the accident progression. RPV pressure is not important (characteristic 5 is N/A) since, even if it occurred, DCH does not significantly affect the source term as the containment does not fail and the vent limits its effect.
8	CD, VB, No CF, N/A, N/A Core damage occurs followed by vessel breach. The containment never fails structurally (characteristic 4 is N/A) and is not vented. RPV pressure is not important (characteristic 5 is N/A) since, even if it occurred, DCH did not fail containment. Some nominal leakage from the containment exists and is accounted for in the analysis so that while the risk will be small it is not completely negligible.
9	CD, No VB, N/A, N/A, N/A Core damage occurs but is arrested in time to prevent vessel breach. There are no releases associated with vessel breach or MCCI. It must be remembered, however, that the containment can fail due to overpressure or venting even if vessel breach is averted. Thus, the potential exists for some of the in-vessel releases to be released to the environment.
10	No CD, N/A, N/A, N/A, N/A Core damage did not occur. No in-vessel or ex-vessel release occurs. The containment may fail on overpressure or be vented. The RPV may be at high or low pressure depending on the progression characteristics. The risk associated with this bin is negligible.

Table 4.2-3
Calculation of PBAPS Population Dose Risk at 50 Miles

Collapsed Bin #	Fractional APB Contributions to Risk (MFCR) (1)	NUREG/CR-4551 Population Dose Risk at 50 miles (From a total of 7.9 person-rem/yr, mean) (2)	NUREG/CR-4551 Collapsed Bin Frequencies (per year) (3)	NUREG/CR-4551 Population Dose at 50 miles (Person-rem) (4)
1	0.021	0.1659	9.55E-08	1.74E+06
2	0.0066	0.05214	4.77E-08	1.09E+06
3	0.556	4.3924	1.48E-06	2.97E+06
4	0.226	1.7854	7.94E-07	2.25E+06
5	0.0022	0.01738	1.30E-08	1.34E+06
6	0.059	0.4661	2.04E-07	2.28E+06
7	0.118	0.9322	4.77E-07	1.95E+06
8	0.0005	0.00395	7.99E-07	4.94E+03
9	0.01	0.079	3.86E-07	2.05E+05
10	0	0	4.34E-08	0
Totals	**1.0**	**7.9**	**4.34E-6**	

(1) Mean Fractional Contribution to Risk from Table 5.2-3 of NUREG/CR-4551

(2) The total population dose risk at 50 miles from internal events in person-rem is provided in Table 5.1-1 of NUREG/CR-4551. The contribution for a given APB is the product of the total PDR50 and the fractional APB contribution.

(3) NUREG/CR-4551 provides the conditional probabilities of the collapsed APBs in Figure 2.5-6. These conditional probabilities are multiplied by the total internal CDF to calculate the collapsed APB frequency.

(4) Obtained from dividing the population dose risk shown in the third column of this table by the collapsed bin frequency shown in the fourth column of this table.

Population Estimate Methodology

The person-rem results in Table 4.2-3 can be used as an approximation of the dose for Limerick if it is corrected for the population surrounding Limerick. The total population within a 50-mile radius of Limerick is projected to be 8.05E+06 by the year 2010 [17]. This value is slightly less than the projected value of 8.53E+06 from the Limerick UFSAR [24] since it factors in more recent actual census data from 1990 and 2000 for the projected growth estimates compared to the earlier population data utilized in the UFSAR. The use of the 2010 estimate is judged to be sufficient to perform this assessment. Scaling the year 2010 population data to the date of the next ILRT test if extended beyond the current due date would not significantly impact the quantitative results, nor would it change the conclusions.

This population value is compared to the population value that is provided in NUREG/CR-4551 in order to get a "Population Dose Factor" that can be applied to the APBs to get dose estimates for Limerick.

Total Limerick Population$_{50miles}$ = 8.05E+06

Peach Bottom Population from NUREG/CR-4551 = 3.02E+06

Population Dose Factor = 8.05E+06 / 3.02E+06 = 2.67

The difference in the doses at 50 miles is assumed to be in direct proportion to the difference in the population within 50 miles of each site. This does not take into account differences in meteorology data, detailed environmental factors or detailed differences in containment designs, but does provide a first-order approximation for Limerick of the population doses associated with each of the release categories from NUREG/CR-4551. This is considered adequate since the conclusions from this analysis will not be substantially affected by the actual dose values that are used.

Table 4.2-4 shows the results of applying the population dose factor to the NUREG/CR-4551 population dose results at 50 miles to obtain the adjusted population dose at 50 miles for Limerick.

Table 4.2-4

Calculation of Limerick Population Dose Risk at 50 Miles

Accident Progression Bin #	NUREG/CR-4551 Population Dose at 50 miles (Person-rem)	Bin Multiplier used to obtain Limerick Population Dose	Limerick Adjusted Population Dose at 50 miles (Person-rem)
1	1.74E+06	2.67	4.65E+06
2	1.09E+06	2.67	2.91E+06
3	2.97E+06	2.67	7.93E+06
4	2.25E+06	2.67	6.01E+06
5	1.34E+06	2.67	3.58E+06
6	2.28E+06	2.67	6.09E+06
7	1.95E+06	2.67	5.21E+06
8	4.94E+03	2.67	1.32E+04
9	2.05E+05	2.67	5.47E+05
10	0	2.67	0.00E+00

Application of Limerick PRA Model Results to NUREG/CR-4551 Level 3 Output

A major factor related to the use of NUREG/CR-4551 in this evaluation is that the results of the Limerick PRA Level 2 model are not defined in the same terms as reported in NUREG/CR-4551. In order to use the Level 3 model presented in that document, it was necessary to apply the Limerick PRA Level 2 model results into a format which allowed for the scaling of the Level 3 results based on current Level 2 output. Finally, as mentioned above, the Level 3 results were modified to reflect the difference in the site demographics that exist between the two sites. This subsection provides a description of the process used to apply the Limerick PRA Level 2 model results into a form that can be used to generate Level 3 results using the NUREG/CR-4551 documentation.

The basic process that was pursued to obtain Level 3 results based on the Limerick Level 2 model and NUREG/CR-4551 was to define a useful relationship between the Level 2 and Level 3 results. Consequently, each non-zero sequence of the Limerick Level 2 model was reviewed and assigned into one of the collapsed Accident Progression Bins (APBs) from NUREG/CR-4551. The Level 2 model contains a significantly larger amount of information about the accident sequences than what is used in the collapsed APBs in NUREG/CR-4551 and this assignment process required simplification of accident progression information and assumptions related to categorizations of certain items. The relevant assumptions used for these assignments are shown in Table 4.2-5. Other nodes are included in the Limerick Level 2 model, but these were not utilized (or did not contribute) to the APB assignment performed here for the ILRT assessment.

Table 4.2-5
Limerick Level 2 Model Nodal Assumptions for Application to the NUREG/CR-4551 Accident Progression Bins

Limerick Containment Event Tree Node	Assumption
IS – Containment Isolation	If the containment is not isolated, it is assumed that it will be open for the equivalent of an un-scrubbed release as soon as the vessel is breached. No depressurization is asked prior to this node; it is assumed that RPV pressure is >= 200 psi for these sequences. This is APB #3.
OP – Operator Depressurizes the RPV	It is assumed that success on this branch results in RPV pressure below 200 psi that is then used to distinguish between APB #1 versus APB #2, or APB #3 versus APB #4.
RX – Core Melt Arrested in Vessel	A success on this branch signifies that there is no vessel breach. The sequences following this path are typically grouped in APB #9. However, this assignment is overridden if the containment still fails due to subsequent CZ or HR-MU failures. In these cases, CZ failures are assigned to APB #3 or APB #4 depending on the status of OP, and APB #6 is assigned for HR-MU failures with CV also failed, but to APB #7 is assigned for HR-MU failures with CV success.
CZ – Containment Intact Early	Failure of containment is assumed to result in an un-scrubbed release and is grouped in APB #3 or APB #4 depending on RPV pressure.
FC and FD – Containment Flooding Initiated and Completed	If containment flooding is initiated and successfully completed without other containment failures, this is assigned to APB #7 based on the interpretation that the successful completion of flooding requires venting.
HR – Containment Heat Removal Maintained	Failure of this branch is assumed to result in a Late DW failure and APB #6 is generally assigned for these sequences. The exception is the case where RX is successful as noted above.
CV – Containment Venting	Success of these nodes is used to indicate assignment to APB #7 for venting as long as the suppression pool is not bypassed in the SP node, and other containment failure nodes are not failed. This assignment applies to sequences with RX failures.

Table 4.2-5
Limerick Level 2 Model Nodal Assumptions for Application to the NUREG/CR-4551 Accident Progression Bins

Limerick Containment Event Tree Node	Assumption
SP – Suppression Pool Not Bypassed	The suppression pool bypass node is considered in the CETs to determine whether the vent volume passes through the suppression pool or not. This node is used to distinguish between a WW or DW failure as described in the other node assumption descriptions above.
MU – Inventory Makeup Available	Success of this node in combination with success of RX is used to assign APB #9 unless otherwise overridden as described in the RX node discussion above.
DI – Drywell Intact / WW – Wetwell Airspace Breach	These nodes were utilized to distinguish between early Drywell Failures (APB #3, APB #4) and Wetwell Failures (APB #1, APB #2). All Late failures were assumed to be drywell failures (APB #6), and therefore no sequences were assigned to APB #5.
RB – Release Mitigated in Reactor Building	The RB node, release mitigated in reactor building, is not credited as a scrubbing mechanism in this analysis. The only scrubbing accounted for in the collapsed bins is distinguished by indicating a WW release (with the success of the SP node).

4.3 Impact of Extension on Detection of Component Failures That Lead to Leakage (Small and Large)

The ILRT can detect a number of component failures such as liner breach, failure of certain bellows arrangements and failure of some sealing surfaces, which can lead to leakage. The proposed ILRT test interval extension may influence the conditional probability of detecting these types of failures. To ensure that this effect is properly accounted for, the EPRI Class 3 accident class as defined in Table 4.1-1 is divided into two sub-classes representing small and large leakage failures. These subclasses are defined as Class 3a and Class 3b, respectively.

The probability of the EPRI Class 3a failures may be determined, consistent with the NEI Guidance [3], as the mean failure estimated from the available data (i.e., 5 "small" failures in 182 tests leads to a 5/182=0.027 mean value). For Class 3b, using the original NEI Guidance [3], a non-informative prior distribution would be assumed for no "large" failures in 182 tests (i.e., 0.5/(182+1) = 0.0027).

In a follow-on letter [21] to their ILRT guidance document [3], NEI issued additional information concerning the potential that the calculated delta LERF values for several plants may fall above the "very small change" guidelines of the NRC regulatory guide 1.174. This additional NEI information includes a discussion of conservatisms in the quantitative guidance for delta LERF. NEI describes ways to demonstrate that, using plant-specific calculations, the delta LERF is smaller than that calculated by the simplified method.

The supplemental information states:

> "The methodology employed for determining LERF (Class 3b frequency) involves conservatively multiplying the CDF by the failure probability for this class (3b) of accident. This was done for simplicity and to maintain conservatism. However, some plant-specific accident classes leading to core damage are likely to include individual sequences that either may already (independently) cause a LERF or could never cause a LERF, and are thus not associated with a postulated large Type A containment leakage path (LERF). These contributors can be removed from Class 3b

> in the evaluation of LERF by multiplying the Class 3b probability by only that portion of CDF that may be impacted by type A leakage."

The application of this additional guidance to the analysis for Limerick, as detailed in Section 5, means that the Class 2 and Class 8 sequences are subtracted from the CDF that is applied to Class 3b. To be consistent, the same change is made to the Class 3a CDF, even though these events are not considered LERF. Class 2 and Class 8 events refer to sequences with either large pre-existing containment isolation failures or containment bypass events. These sequences are already considered to contribute to LERF in the Limerick Level 2 PRA analysis.

Consistent with the NEI Guidance [3], the change in the leak detection probability can be estimated by comparing the average time that a leak could exist without detection. For example, the average time that a leak could go undetected with a three-year test interval is 1.5 years (3 yr / 2), and the average time that a leak could exist without detection for a ten-year interval is 5 years (10 yr / 2). This change would lead to a non-detection probability that is a factor of 3.33 (5.0/1.5) higher for the probability of a leak that is detectable only by ILRT testing, given a 10-year vs. a 3-yr interval. Correspondingly, an extension of the ILRT interval to fifteen years can be estimated to lead to about a factor of 5.0 (7.5/1.5) increase in the non-detection probability of a leak.

It should be noted that using the methodology discussed above is very conservative compared to previous submittals (e.g., the IP3 request for a one-time ILRT extension that was approved by the NRC [6]) because it does not factor in the possibility that the failures could be detected by other tests (e.g., the Type B local leak rate tests that will still occur.) Eliminating this possibility conservatively over-estimates the factor increases attributable to the ILRT extension.

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

An estimate of the likelihood and risk implications of corrosion-induced leakage of the steel liners occurring and going undetected during the extended test interval is evaluated using the methodology from the Calvert Cliffs liner corrosion analysis [19]. The Calvert Cliffs analysis was performed for a concrete cylinder and dome and a concrete basemat, each with a steel liner. The Limerick primary containment is a pressure-suppression BWR/Mark II containment type that also includes a steel-lined reinforced concrete structure.

The following approach is used to determine the change in likelihood, due to extending the ILRT, of detecting corrosion of the containment steel liner. This likelihood is then used to determine the resulting change in risk. Consistent with the Calvert Cliffs analysis, the following issues are addressed:

- Differences between the containment basemat and the containment walls
- The historical steel liner flaw likelihood due to concealed corrosion
- The impact of aging
- The corrosion leakage dependency on containment pressure
- The likelihood that visual inspections will be effective at detecting a flaw

Assumptions

- Consistent with the Calvert analysis, a half failure is assumed for basemat concealed liner corrosion due to the lack of identified failures. (See Table 4.4-1, Step 1.)

- The two corrosion events used to estimate the liner flaw probability in the Calvert Cliffs analysis are assumed to be applicable to the Limerick containment analysis. These events, one at North Anna Unit 2 and one at Brunswick Unit 2, were initiated from the non-visible (backside) portion of the containment liner.

- For consistency with the Calvert Cliffs analysis, the estimated historical flaw probability is limited to 5.5 years to reflect the years since September 1996 when 10 CFR 50.55a started requiring visual inspection. Additional success data were not used to limit the aging impact of this corrosion issue, even though inspections were being performed prior to this date (and have been performed since the time frame of the Calvert Cliffs analysis), and there is no evidence that additional corrosion issues were identified. (See Table 4.4-1, Step 1.)

- Consistent with the Calvert Cliffs analysis, the steel liner flaw likelihood is assumed to double every five years. This is based solely on judgment and is included in this analysis to address the increased likelihood of corrosion as the steel liner ages. (See Table 4.4-1, Steps 2 and 3.) Sensitivity studies are included that address doubling this rate every ten years and every two years.

- In the Calvert Cliffs analysis, the likelihood of the containment atmosphere reaching the outside atmosphere given that a liner flaw exists was estimated as 1.1% for the containment walls and dome region and 0.11% (10% less) for the basemat. These values were determined from an assessment of the containment fragility curve versus the ILRT test pressure. For Limerick the containment failure probabilities are conservatively assumed to be 10% for the drywell and wetwell outer walls, and since the basemat for the Limerick Mark II containment is in the suppression pool, it is judged that failure of this area would not lead to LERF. In any event, a 1% probability is conservatively assigned. Sensitivity studies are included that increase and decrease the probabilities by an order of magnitude. (See Table 4.4-1, Step 4.)

- Consistent with the Calvert analysis, a 5% visual inspection detection failure likelihood given the flaw is visible and a total detection failure likelihood of 10% is used. To date, all liner corrosion events have been detected through visual inspection. (See Table 4.4-1, Step 5.) Sensitivity studies are included that evaluate total detection failure likelihood of 5% and 15%, respectively.

- Consistent with the Calvert analysis, all non-detectable containment failures are assumed to result in early releases. This approach avoids a detailed analysis of containment failure timing and operator recovery actions.

Table 4.4-1
Steel Liner Corrosion Base Case

Step	Description	Containment Wall	Containment Basemat
1	**Historical Steel Liner Flaw Likelihood** Failure Data: Containment location specific (consistent with Calvert Cliffs analysis).	Events: 2 2/(70 * 5.5) = **5.2E-3**	Events: 0 (assume half a failure) 0.5/(70 * 5.5) = **1.3E-3**
2	**Age Adjusted Steel Liner Flaw Likelihood** During 15-year interval, assume failure rate doubles every five years (14.9% increase per year). The average for 5th to 10th year is set to the historical failure rate (consistent with Calvert Cliffs analysis).	Year — Failure Rate 1 — 2.1E-3 avg 5-10 — 5.2E-3 15 — 1.4E-2 **15 year average = 6.27E-3**	Year — Failure Rate 1 — 5.0E-4 avg 5-10 — 1.3E-3 15 — 3.5E-3 **15 year average = 1.57E-3**
3	**Flaw Likelihood at 3, 10, and 15 years** Uses age adjusted liner flaw likelihood (Step 2), assuming failure rate doubles every five years (consistent with Calvert Cliffs analysis – See Table 6 of Reference [19]).	**0.71% (1 to 3 years)** **4.06% (1 to 10 years)** **9.40% (1 to 15 years)** (Note that the Calvert Cliffs analysis presents the delta between 3 and 15 years of 8.7% to utilize in the estimation of the delta-LERF value. For this analysis the values are calculated based on the 3, 10, and 15 year intervals.)	**0.18% (1 to 3 years)** **1.02% (1 to 10 years)** **2.35% (1 to 15 years)** (Note that the Calvert Cliffs analysis presents the delta between 3 and 15 years of 2.2% to utilize in the estimation of the delta-LERF value. For this analysis the values are calculated based on the 3, 10, and 15 year intervals.)
4	**Likelihood of Breach in Containment Given Steel Liner Flaw** The failure probability of the containment is assumed to be 10% (compared to 1.1% in the Calvert Cliffs analysis). The basemat failure probability is assumed to be a factor of ten less, 1%, (compared to 0.11% in the Calvert Cliffs analysis).	**10%**	**1%**

Table 4.4-1
Steel Liner Corrosion Base Case

Step	Description	Containment Wall	Containment Basemat
5	**Visual Inspection Detection Failure Likelihood** Utilize assumptions consistent with Calvert Cliffs analysis.	**10%** 5% failure to identify visual flaws plus 5% likelihood that the flaw is not visible (not through-cylinder but could be detected by ILRT). All events have been detected through visual inspection. 5% visible failure detection is a conservative assumption.	**100%** Cannot be visually inspected.
6	**Likelihood of Non-Detected Containment Leakage** (Steps 3 * 4* 5)	**0.0071% (at 3 years)** 0.71% * 10% * 10% **0.0406% (at 10 years)** 4.06% * 10% * 10% **0.0940% (at 15 years)** 9.40% * 10% * 10%	**0.0018% (at 3 years)** 0.18% * 1% * 100% **0.0102% (at 10 years)** 1.02% * 1% * 100% **0.0235% (at 15 years)** 2.35% * 1% * 100%

The total likelihood of the corrosion-induced, non-detected containment leakage is the sum of Step 6 for the containment walls and the containment basemat:

At 3 years: 0.0071% + 0.0018% = 0.0089%

At 10 years: 0.0406% + 0.0102% = 0.0508%

At 15 years: 0.0940% + 0.0235% = 0.1175%

<u>Limerick Past ILRT Results</u>

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.

Based on completion of two successful ILRTs at each of the Liemrick units, the current ILRT interval is once per ten years. The next Type A test for Limerick Unit 1 is currently due to be completed by 3/2008, and by 3/2009 for Unit 2 [18].

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 portion of Limerick specific information utilized is the fact that the current ILRT interval is once per ten years.

<u>NEI Interim Guidance</u>

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 for Limerick, 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, referred to as the conditional containment failure probability, CCFP. The NRC has previously accepted similar calculations [7] for the change in 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.

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).

5.0 RESULTS

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, 7, 20, 23] have led to the following results. The results are displayed according to the eight accident classes defined in the EPRI report. Table 5.0-1 lists these accident classes.

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

- Core damage sequences in which the containment remains intact initially and in the long term (EPRI TR-104285 Class 1 sequences).
- Core damage sequences in which containment integrity is impaired due to random isolation failures of plant components other than those associated with Type B or Type C test components. For example, liner breach or bellows leakage. (EPRI TR-104285 Class 3 sequences).
- Core damage sequences in which containment integrity is impaired due to containment isolation failures of pathways left "opened" following a plant post-maintenance test. (For example, a valve failing to close following a valve stroke test. (EPRI TR-104285 Class 6 sequences). Consistent with the NEI Guidance, this class is not specifically examined since it will not significantly influence the results of this analysis.
- Accident sequences involving containment bypassed (EPRI TR-104285 Class 8 sequences), large containment isolation failures (EPRI TR-104285 Class 2 sequences), and small containment isolation "failure-to-seal" events (EPRI TR-104285 Class 4 and 5 sequences) are accounted for in this evaluation as part of the baseline risk profile. However, they are not affected by the ILRT frequency change.
- Class 4 and 5 sequences are impacted by changes in Type B and C test intervals; therefore, changes in the Type A test interval do not impact these sequences.

Table 5.0-1

ACCIDENT CLASSES

Accident Classes (Containment Release Type)	Description
1	No Containment Failure
2	Large Isolation Failures (Failure to Close)
3a	Small Isolation Failures (liner breach)
3b	Large Isolation Failures (liner breach)
4	Small Isolation Failures (Failure to seal –Type B)
5	Small Isolation Failures (Failure to seal—Type C)
6	Other Isolation Failures (e.g., dependent failures)
7	Failures Induced by Phenomena (Early and Late)
8	Bypass (SGTR and Interfacing System LOCA)
CDF	All CET End states (including very low and no release)

The steps taken to perform this risk assessment evaluation are as follows:

Step 1 - Quantify the base-line risk in terms of frequency per reactor year for each of the eight accident classes presented in Table 5.0-1.

Step 2 - Develop plant-specific person-rem dose (population dose) per reactor year for each of the eight accident classes.

Step 3 - Evaluate risk impact of extending Type A test interval from 3 to 15 and 10 to 15 years.

Step 4 - Determine the change in risk in terms of Large Early Release Frequency (LERF) in accordance with RG 1.174.

Step 5 - Determine the impact on the Conditional Containment Failure Probability (CCFP)

5.1 Step 1 – Quantify the Base-Line Risk in Terms of Frequency per Reactor Year

This step involves the review of the Limerick containment event trees (CETs) and Level 2 accident sequence frequency results. The CETs characterize the response of the containment to important severe accident sequences. As described in Section 4.2, the Limerick CETs were examined and each endstate was applied to one of the Accident Progression Bins as defined in NUREG/CR-4551. The correlation between the NUREG/CR-4551 Accident Progression Bins to the EPRI containment release categories is shown in Table 5.1-1. This application combined with the Limerick dose (person-rem) results determined from Table 4.2-4 forms the basis for estimating the population dose for Limerick.

As previously described, the extension of the Type A interval does not influence those accident progressions that involve large containment isolation failures, Type B or Type C testing, or containment failure induced by severe accident phenomena.

For the assessment of ILRT impacts on the risk profile, the potential for pre-existing leaks is included in the model. (These events are represented by the Class 3 sequences in EPRI TR-104285). Two failure modes were considered for the Class 3 sequences. These are Class 3a (small breach) and Class 3b (large breach).

The frequencies for the severe accident classes defined in Table 5.0-1 were developed for Limerick based on the assumptions shown in Table 4.2-5, determining the frequencies for Classes 3a and 3b, and then determining the remaining frequency for Class 1. Furthermore, adjustments were made to the Class 3b and hence Class 1 frequencies to account for the impact of undetected corrosion of the steel liner per the methodology described in Section 4.4. The eight containment release class frequencies were developed consistent with the definitions in Table 5.0-1 as described following Table 5.1-1.

Table 5.1-1

Containment Release Type Assignment from the NUREG/CR-4551 Consequence Model

EPRI TR-104285 Containment Release		NUREG/CR-4551	
SCENARIO TYPE	**Dose (Person-Rem)**	**Accident Progression Bin**	**Limerick Dose (Person-Rem)**
1	1.32E+04 (1)	8 (VB, No CF, No Vent)	1.32E+04
		10 (No core damage)	0.00E+00
2 (1)	7.93E+06	3 (VB, Early DW, Hi Press)	7.93E+06
7	3.97E+6 (2)	1 (VB, Early WW, Hi Press)	4.65E+06
		2 (VB, Early WW, Lo Press)	2.91E+06
		5 (VB, Late WW)	3.58E+06
		6 (VB, Late DW)	6.09E+06
		7 (VB, No CF, Vent)	5.21E+06
		9 (No VB, No CF, No Vent)	5.47E+05
8 (1)	6.01E+6	4 (VB, Early DW, Lo Press)	6.01E+06

(1) No specific Release Bin for this category exists in NUREG/CR-4551. For simplicity, all sequences assigned to APB #3 is used in this analysis to represent EPRI Class 2 and all sequences assigned to APB #4 are assigned to EPRI Class 8. This will not impact the calculated change for the proposed ILRT extension.

(2) Given that multiple NUREG/CR-4551 discrete scenarios apply to the broader EPRI type, the EPRI type dose is based on a weighted average (weights based on Limerick PRA scenario frequencies) of the applicable NUREG/CR-4551 APB doses.

Class 1 Sequences. This group consists of all core damage accident progression bins for which the containment remains intact (modeled as Technical Specification Leakage). The frequency per year for these sequences is 1.22E-06/yr and is determined by subtracting all containment failure end states including the EPRI/NEI Class 3a and 3b frequency calculated below, from the total CDF. For this analysis, the associated maximum containment leakage for this group is $1L_a$, consistent with an intact containment evaluation.

Class 2 Sequences. This group consists of all core damage accident progression bins for which a failure to isolate the containment occurs. For simplicity, the frequency is obtained from all sequences that were assigned to APB #3 for Limerick which is 2.18E-08/yr.

Class 3 Sequences. This group consists of all core damage accident progression bins for which a pre-existing leakage in the containment structure (e.g., containment liner) exists. The containment leakage for these sequences can be either small ($2L_a$ to $35L_a$) or large ($>35L_a$).

The respective frequencies per year are determined as follows:

$PROB_{Class\_3a}$ = probability of small pre-existing containment liner leakage
= 0.027 [see Section 4.3]
$PROB_{Class\_3b}$ = probability of large pre-existing containment liner leakage
= 0.0027 [see Section 4.3]

As described in section 4.3, additional consideration is made to not apply these failure probabilities on those cases that are already LERF scenarios (i.e., the Class 2 and Class 8 contributions).

Class_3a = 0.027 ∗ (CDF-Class 2-Class 8)
= 0.027 * (3.70E-06 – 2.18E-08 – 4.65E-08) = 9.80E-08/yr

Class_3b = 0.0027 ∗ (CDF-Class 2-Class 8)
= 0.0027 * (3.70E-06 – 2.18E-08 – 4.65E-08) = 9.80E-09/yr

For this analysis, the associated containment leakage for Class 3a is $10L_a$ and for Class 3b is $35L_a$. These assignments are consistent with the NEI Interim Guidance.

Class 4 Sequences. This group consists of all core damage accident progression bins for which containment isolation failure-to-seal of Type B test components occurs. Because these failures are detected by Type B tests which are unaffected by the Type A ILRT, this group is not evaluated any further in the analysis.

Class 5 Sequences. This group consists of all core damage accident progression bins for which a containment isolation failure-to-seal of Type C test components. Because the failures are detected by Type C tests which are unaffected by the Type A ILRT, this group is not evaluated any further in this analysis.

Class 6 Sequences. This group is similar to Class 2. These are sequences that involve core damage accident progression bins for which a failure-to-seal containment leakage due to failure to isolate the containment occurs. These sequences are dominated by misalignment of containment isolation valves following a test/maintenance evolution. Consistent with the NEI Interim Guidance, however, this accident class is not explicitly considered since it has a negligible impact on the results.

Class 7 Sequences. This group consists of all core damage accident progression bins in which containment failure induced by severe accident phenomena occurs. For this analysis, the associated radionuclide releases are based on the application of the Level 2 endstates to the Accident Progression Bins from NUREG/CR-4551 as described in Section 4.2. The Class 7 Sequences are divided into 6 categories which consists of Bins 1, 2, 5, 6, 7, and 9 from NUREG/CR-4551. The failure frequency and population dose for each specific APB is shown below in Table 5.1-2. The total release frequency and total dose are then used to determine a weighted average person-rem for use as the representative EPRI Class 7 dose in the subsequent analysis. Note that the total frequency and dose associated from this EPRI class does not change as part of the ILRT extension request.

Table 5.1-2
ACCIDENT CLASS 7 FAILURE FREQUENCIES AND POPULATION DOSES (LIMERICK BASE CASE LEVEL 2 MODEL)

Accident Class (APB umber)	Release Frequency/yr	Population Dose (50 miles) Person-Rem (1)	Population Dose Risk (50 Miles) (Person-Rem/yr) (2)
7a (APB #1)	7.40E-09	4.65E+06	3.44E-02
7b (APB #2)	2.39E-07	2.91E+06	6.97E-01
7c (APB #5)	0.00E+00	3.58E+06	0.00E+00
7d (APB #6)	7.76E-08	6.09E+06	4.72E-01
7e (APB #7)	1.47E-06	5.21E+06	7.67E+00
7f (APB #9)	5.09E-07	5.47E+05	2.78E-01
Class 7 Total	**2.31E-06**	**3.97E+06** (3)	**9.15E+00**

(1) Population dose values obtained from Table 4.2-4 based on the Accident Progression Bin.
(2) Obtained by multiplying the Release Frequency value from the second column of this table by the Population dose value from the third column of this table.
(3) The weighted average population dose for Class 7 is obtained by dividing the total population dose risk by the total release frequency.

Class 8 Sequences. This group consists of all core damage accident progression bins in which containment bypass occurs. For simplicity, the frequency is obtained from all sequences that were assigned to APB #4 for Limerick which is 4.65E-08/yr.

Summary of Accident Class Frequencies

In summary, the accident sequence frequencies that can lead to radionuclide release to the public have been derived consistent with the definition of Accident Classes defined in EPRI-TR-104285. Table 5.1-3 summarizes these accident frequencies by Accident Class.

Table 5.1-3
RADIONUCLIDE RELEASE FREQUENCIES AS A FUNCTION OF ACCIDENT CLASS (LIMERICK BASE CASE)

Accident Classes (Containment Release Type)	Description	Frequency (per Rx-yr)
1	No Containment Failure	1.22E-06
2[(1)]	Large Isolation Failures (Failure to Close)	2.18E-08
3a	Small Isolation Failures (liner breach)	9.80E-08
3b	Large Isolation Failures (liner breach)	9.80E-09
4	Small Isolation Failures (Failure to seal –Type B)	N/A
5	Small Isolation Failures (Failure to seal—Type C)	N/A
6	Other Isolation Failures (e.g., dependent failures)	N/A
7	Failures Induced by Phenomena (Early and Late)	2.31E-06
8[(1)]	Bypass (Interfacing System LOCA)	4.65E-08
CDF	**All CET End states (including very low and no release)**	**3.70E-06**

(1) The EPRI Class 2 and Class 8 scenarios are assumed to be LERF in the ILRT methodology, and the sum of these sequence contributions from the simplified APB assignment of 6.83E-08/yr agrees quite well with the Limerick detailed Level 2 PRA model reported LERF value of 6.80E-08/yr.

5.2 Step 2 – Develop Plant-Specific Person-Rem Dose (Population Dose) per Reactor Year

Plant-specific release analyses were performed to estimate the person-rem doses to the population within a 50-mile radius from the plant. The releases are based on information provided by NUREG/CR-4551 with adjustments made for the site demographic differences compared to the reference plant as described in Section 4.2, and summarized in Table 4.2-4. The results of applying these releases to the EPRI/NEI containment failure classification are as follows:

Class 1	=	1.32E+04 person-rem (at $1.0L_a$) [(1)]
Class 2	=	7.93E+06[(2)]
Class 3a	=	1.32E+04 person-rem x $10L_a$ = 1.32E+05 person-rem [(3)]
Class 3b	=	1.32E+04 person-rem x $35L_a$ = 4.62E+05 person-rem [(3)]
Class 4	=	Not analyzed
Class 5	=	Not analyzed
Class 6	=	Not analyzed
Class 7	=	3.97E+06 person-rem [(4)]
Class 8	=	6.01E+06 person-rem [(5)]

(1) The Class 1, containment intact sequences, dose is assigned from the APB #8 (No CF, No Vent) from the NUREG/CR-4551 adjusted dose for Limerick as shown in Table 4.2-4.

(2) The Class 2, containment isolation failures, dose is approximated from APB #3 (VB, Early DW, Hi Press) from Table 4.2-4.

(3) The Class 3a and 3b dose are related to the leakage rate as shown. This is consistent with the NEI Interim Guidance.

(4) The Class 7 dose is assigned from the weighted average dose calculated from APBs #1, 2, 5, 6, 7, and 9 from Table 4.2-4 as detailed in Table 5.1-2 above.

(5) Class 8 sequences involve containment bypass failures; as a result, the person-rem dose is not based on normal containment leakage. As an approximation, the releases for this class are assigned from APB #4 (VB, Early DW, Lo Press) from Table 4.2-4.

In summary, the population dose estimates derived for use in the risk evaluation per the EPRI methodology [2] containment failure classifications, and consistent with the NEI guidance [3] are provided in Table 5.2-1.

Table 5.2-1
LIMERICK POPULATION DOSE ESTIMATES
FOR POPULATION WITHIN 50 MILES

Accident Classes (Containment Release Type)	Representative Accident Progression Bin (APB)	Description	Person-Rem (50 miles)
1	8	No Containment Failure (1 La)	1.32E+04
2	3	Large Isolation Failures (Failure to Close)	7.93E+06
3a	$10L_a$	Small Isolation Failures (liner breach)	1.32E+05
3b	$35L_a$	Large Isolation Failures (liner breach)	4.62E+05
4	N/A	Small Isolation Failures (Failure to seal –Type B)	NA
5	N/A	Small Isolation Failures (Failure to seal—Type C)	NA
6	N/A	Other Isolation Failures (e.g., dependent failures)	NA
7	1, 2, 5, 6, 7, 9	Failures Induced by Phenomena (Early and Late)	3.97E+06
8	4	Bypass (SGTR and Interfacing System LOCA)	6.01E+06

The above dose estimates, when combined with the frequency results presented in Table 5.1-3, yield the Limerick baseline mean consequence measures for each accident class. These results are presented in Table 5.2-2.

Table 5.2-2

LIMERICK ANNUAL DOSE AS A FUNCTION OF ACCIDENT CLASS; CHARACTERISTIC OF CONDITIONS FOR ILRT REQUIRED 3/10 YEARS

Accident Classes (Containment Release Type)	Description	Person-Rem (50 miles)	NEI Methodology		NEI Methodology Plus Corrosion		Change Due to Corrosion Person-Rem/yr(1)
			Frequency (per Rx-yr)	Person-Rem/yr (50 miles)	Frequency (per Rx-yr)	Person-Rem/yr (50 miles)	
1	No Containment Failure (2)	1.32E+04	1.22E-06	1.60E-02	1.22E-06	1.60E-02	-4.25E-06
2	Large Isolation Failures (Failure to Close)	7.93E+06	2.18E-08	1.73E-01	2.18E-08	1.73E-01	--
3a	Small Isolation Failures (liner breach)	1.32E+05	9.80E-08	1.29E-02	9.80E-08	1.29E-02	--
3b	Large Isolation Failures (liner breach)	4.62E+05	9.80E-09	4.52E-03	1.01E-08	4.67E-03	1.49E-04
4	Small Isolation Failures (Failure to seal –Type B)	NA	N/A	N/A	N/A	N/A	N/A
5	Small Isolation Failures (Failure to seal—Type C)	NA	N/A	N/A	N/A	N/A	N/A
6	Other Isolation Failures (e.g., dependent failures)	NA	N/A	N/A	N/A	N/A	N/A
7	Failures Induced by Phenomena (Early and Late)	3.97E+06	2.31E-06	9.15E+00	2.31E-06	9.15E+00	--
8	Bypass (SGTR and ISLOCA)	6.01E+06	4.65E-08	2.79E-01	4.65E-08	2.79E-01	--
CDF	All CET end states		3.70E-06	9.63	3.70E-06	9.63	1.44E-04

1) Only release Classes 1 and 3b are affected by the corrosion analysis.

2) Characterized as $1L_a$ release magnitude consistent with the derivation of the ILRT non-detection failure probability for ILRTs. Release classes 3a and 3b include failures of containment to meet the Technical Specification leak rate.

The calculated dose for Limerick compares favorably with other locations given the relative population densities surrounding each location:

Plant	Annual Dose (Person-Rem/Yr)	Reference
Indian Point 3	14,515	[6]
Peach Bottom	6.2	[23]
Crystal River	1.4	[20]
Limerick	9.6	[Table 5.2-2]

5.3 Step 3 – Evaluate Risk Impact of Extending Type A Test Interval From 10-to-15 Years

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

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

As previously stated, Type A tests impact only Class 3 sequences. For Class 3 sequences, the release magnitude is not impacted by the change in test interval (a small or large breach remains the same, even though the probability of not detecting the breach increases). Thus, only the frequency of Class 3a and 3b sequences is impacted. The risk contribution is changed based on the NEI guidance as described in Section 4.3 by a factor of 3.33 compared to the base case values. The results of the calculation for a 10-year interval are presented in Table 5.3-1 for Limerick.

<u>Risk Impact Due to 15-Year Test Interval</u>

The risk contribution for a 15-year interval is calculated in a manner similar to the 10-year interval. The difference is in the increase in probability of leakage in Classes 3a and 3b. For this case, the value used in the analysis is a factor of 5.0 compared to the 3-year interval value, as described in Section 4.3. The results for this calculation are presented in Table 5.3-2.

Table 5.3-1

LIMERICK ANNUAL DOSE AS A FUNCTION OF ACCIDENT CLASS; CHARACTERISTIC OF CONDITIONS FOR ILRT REQUIRED 1/10 YEARS

Accident Classes (Containment Release Type)	Description	Person-Rem (50 miles)	NEI Methodology		NEI Methodology Plus Corrosion		Change Due to Corrosion Person-Rem/yr[1]
			Frequency (per Rx-yr)	Person-Rem/yr (50 miles)	Frequency (per Rx-yr)	Person-Rem/yr (50 miles)	
1	No Containment Failure [2]	1.32E+04	9.65E-07	1.27E-02	9.63E-07	1.27E-02	-2.43E-05
2	Large Isolation Failures (Failure to Close)	7.93E+06	2.18E-08	1.73E-01	2.18E-08	1.73E-01	--
3a	Small Isolation Failures (liner breach)	1.32E+05	3.26E-07	4.30E-02	3.26E-07	4.30E-02	--
3b	Large Isolation Failures (liner breach)	4.62E+05	3.26E-08	1.51E-02	3.45E-08	1.59E-02	8.50E-04
4	Small Isolation Failures (Failure to seal – Type B)	NA	N/A	N/A	N/A	N/A	N/A
5	Small Isolation Failures (Failure to seal—Type C)	NA	N/A	N/A	N/A	N/A	N/A
6	Other Isolation Failures (e.g., dependent failures)	NA	N/A	N/A	N/A	N/A	N/A
7	Failures Induced by Phenomena (Early and Late)	3.97E+06	2.31E-06	9.15E+00	2.31E-06	9.15E+00	--
8	Bypass (SGTR and ISLOCA)	6.01E+06	4.65E-08	2.79E-01	4.65E-08	2.79E-01	--
CDF	All CET end states		3.70E-06	9.67	3.70E-06	9.67	8.26E-04

[1] Only release classes 1 and 3b are affected by the corrosion analysis.

[2] Characterized as $1L_a$ release magnitude consistent with the derivation of the ILRT non-detection failure probability for ILRTs. Release classes 3a and 3b include failures of containment to meet the Technical Specification leak rate.

Table 5.3-2

LIMERICK ANNUAL DOSE AS A FUNCTION OF ACCIDENT CLASS; CHARACTERISTIC OF CONDITIONS FOR ILRT REQUIRED 1/15 YEARS

Accident Classes (Containment Release Type)	Description	Person-Rem (50 miles)	NEI Methodology		NEI Methodology Plus Corrosion		Change Due to Corrosion Person-Rem/yr(1)
			Frequency (per Rx-yr)	Person-Rem/yr (50 miles)	Frequency (per Rx-yr)	Person-Rem/yr (50 miles)	
1	No Containment Failure (2)	1.32E+04	7.85E-07	1.03E-02	7.80E-07	1.03E-02	-5.63E-05
2	Large Isolation Failures (Failure to Close)	7.93E+06	2.18E-08	1.73E-01	2.18E-08	1.73E-01	--
3a	Small Isolation Failures (liner breach)	1.32E+05	4.900E-07	6.46E-02	4.90E-07	6.46E-02	--
3b	Large Isolation Failures (liner breach)	4.62E+05	4.900E-08	2.26E-02	5.33E-08	2.46E-02	1.97E-03
4	Small Isolation Failures (Failure to seal – Type B)	NA	N/A	N/A	N/A	N/A	N/A
5	Small Isolation Failures (Failure to seal—Type C)	NA	N/A	N/A	N/A	N/A	N/A
6	Other Isolation Failures (e.g., dependent failures)	NA	N/A	N/A	N/A	N/A	N/A
7	Failures Induced by Phenomena (Early and Late)	3.97E+06	2.31E-06	9.15E+00	2.31E-06	9.15E+00	--
8	Bypass (SGTR and ISLOCA)	6.01E+06	4.65E-08	2.79E-01	4.65E-08	2.79E-01	--
CDF	All CET end states		3.70E-06	9.70	3.70E-06	9.70	1.91E-03

(1) Only release classes 1 and 3b are affected by the corrosion analysis.

(2) Characterized as $1L_a$ release magnitude consistent with the derivation of the ILRT non-detection failure probability for ILRTs. Release classes 3a and 3b include failures of containment to meet the Technical Specification leak rate.

5.4 Step 4 – Determine the Change in Risk in Terms of Large Early Release Frequency

Regulatory Guide 1.174 provides guidance for determining the risk impact of plant-specific changes to the licensing basis. RG 1.174 defines very small changes in risk as resulting in increases of core damage frequency (CDF) below 10^{-6}/yr and increases in LERF below 10^{-7}/yr, and small changes in LERF as below 10^{-6}/yr. Because the ILRT does not impact CDF, the relevant metric is LERF.

For Limerick, 100% of the frequency of Class 3b sequences can be used as a conservative first-order estimate to approximate the potential increase in LERF from the ILRT interval extension (consistent with the NEI guidance methodology). Based on the original 3/10 year test interval assessment from Table 5.2-2, the Class 3b frequency is 1.01E-08/yr. Based on a ten-year test interval from Table 5.3-1, the Class 3b frequency is 3.45E-08/yr; and, based on a fifteen-year test interval from Table 5.3-2, it is 5.33E-08/yr. Thus, the increase in the overall probability of LERF due to Class 3b sequences that is due to increasing the ILRT test interval from 3 to 15 years is 4.32E-08/yr. Similarly, the increase due to increasing the interval from 10 to 15 years is 1.88E-08/yr. As can be seen, even with the conservatisms included in the evaluation (per the NEI methodology), the estimated change in LERF is below the threshold criteria for a very small change in risk when comparing the 15 year results to the current 10-year requirement or to the original 3-in-10 year requirement.

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

Another parameter that the NRC guidance in RG 1.174 states can provide input into the decision-making process is the 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 CCFP can be calculated from the results of this analysis. One of the difficult aspects of this calculation is providing a definition of the "failed containment." In this assessment, the CCFP is defined such that containment failure includes all radionuclide release end states other than the intact state. The conditional part of the definition is conditional given a severe accident (i.e., core damage).

The change in CCFP can be calculated by using the method specified in the NEI Interim Guidance. The NRC has previously accepted similar calculations [7] as the basis for showing that the proposed change is consistent with the defense-in-depth philosophy.

CCFP 3 in 10 yrs	**CCFP 1 in 10 yrs**	**CCFP 1 in 15 yrs**	**ΔCCFP$_{15\text{-}3}$**	**ΔCCFP$_{15\text{-}10}$**
64.48%	65.14%	65.65%	1.17%	0.51%

CCFP = [1 - (Class 1 frequency + Class 3a frequency) / CDF] * 100%

The change in CCFP of slightly more than 1% by extending the test interval to 15 years from the original 3-in-10 year requirement is judged to be insignificant.

5.6 Summary of Results

The results from this ILRT extension risk assessment for Limerick are summarized in Table 5.6-1.

Table 5.6-1

Limerick ILRT Cases: Base, 3 to 10, and 3 to 15 Yr Extensions (Including Age Adjusted Steel Liner Corrosion Likelihood)

EPRI Class	DOSE Per-Rem	Base Case 3 in 10 Years		Extend to 1 in 10 Years		Extend to 1 in 15 Years	
		CDF/Yr	Per-Rem/Yr	CDF/Yr	Per-Rem/Yr	CDF/Yr	Per-Rem/Yr
1	1.32E+04	1.22E-06	1.60E-02	9.63E-07	1.27E-02	7.80E-07	1.03E-02
2	7.93E+06	2.18E-08	1.73E-01	2.18E-08	1.73E-01	2.18E-08	1.73E-01
3a	1.32E+05	9.80E-08	1.29E-02	3.26E-07	4.30E-02	4.90E-07	6.46E-02
3b	4.62E+05	1.01E-08	4.67E-03	3.45E-08	1.59E-02	5.33E-08	2.46E-02
7	3.97E+06	2.31E-06	9.15E+00	2.31E-06	9.15E+00	2.31E-06	9.15E+00
8	6.01E+06	4.65E-08	2.79E-01	4.65E-08	2.79E-01	4.65E-08	2.79E-01
Total		3.70E-06	9.63	3.70E-06	9.67	3.70E-06	9.70
ILRT Dose Rate from 3a and 3b		1.76E-02		5.90E-02		8.92E-02	
Delta Total Dose Rate	From 3 yr	---		3.80E-02		6.59E-02	
	From 10 yr	---		---		2.79E-02	
3b Frequency (LERF)		1.01E-08		3.45E-08		5.33E-08	
Delta LERF	From 3 yr	---		2.44E-08		4.31E-08	
	From 10 yr	---		---		1.88E-08	
CCFP %		64.48%		65.14%		65.65%	
Delta CCFP %	From 3 yr	---		0.66%		1.17%	
	From 10 yr	---		---		0.51%	

6.0 SENSITIVITIES

6.1 Sensitivity to Corrosion Impact Assumptions

The results in Tables 5.3-1, 5.3-2, and 6.1-1 show that including corrosion effects calculated using the assumptions described in Section 4.4 does not significantly affect the results of the ILRT extension risk assessment.

Sensitivity cases were developed to gain an understanding of the sensitivity of the results to the key parameters in the corrosion risk analysis. The time for the flaw likelihood to double was adjusted from every five years to every two and every ten years. The failure probabilities for the containment wall and basemat were increased and decreased by an order of magnitude. The total detection failure likelihood was adjusted from 10% to 15% and 5%. The results are presented in Table 6.1-1. In every case the impact from including the corrosion effects is very minimal. Even the upper bound estimates with very conservative assumptions for all of the key parameters yield increases in LERF due to corrosion of only 1.26E-7/yr. The results indicate that even with very conservative assumptions, the conclusions from the base analysis would not change.

Table 6.1-1
Steel Liner Corrosion Sensitivity Cases

Age (Step 3 in the corrosion analysis)	Containment Breach (Step 4 in the corrosion analysis)	Visual Inspection & Non-Visual Flaws (Step 5 in the corrosion analysis)	Increase in Class 3b Frequency (LERF) for ILRT Extension 3 to 15 years (per Rx-yr)	
			Total Increase	**Increase Due to Corrosion**
Base Case Doubles every 5 yrs	Base Case (10% Wall, 1.0% Basemat)	Base Case 10% Wall, 100% Basemat	4.31E-08	3.94E-09
Doubles every 2 yrs	Base	Base	4.82E-08	9.01E-09
Doubles every 10 yrs	Base	Base	4.25E-08	3.32E-09
Base	Base	15% Wall	4.47E-08	5.52E-09
Base	Base	5% Wall	4.16E-08	2.37E-09
Base	100% Wall, 10% Basemat	Base	7.86E-08	3.94E-08
Base	1.0% Wall, 0.1% Basemat	Base	3.96E-08	3.94E-10
Lower Bound				
Doubles every 10 yrs	1.0% Wall, 0.1% Basemat	5% Wall 100% Basemat	3.94E-08	1.99E-10
Upper Bound				
Doubles every 2 yrs	100% Wall, 10% Basemat	15% Wall 100% Basemat	1.65E-07	1.26E-07

6.2 EPRI Expert Elicitation Sensitivity

An expert elicitation was performed to reduce excess conservatisms in the data associated with the probability of undetected leaks within containment [22]. Since the risk impact assessment of the extensions to the ILRT interval is sensitive to both the probability of the leakage as well as the magnitude, it was decided to perform the expert elicitation in a manner to solicit the probability of leakage as a function of leakage magnitude. In addition, the elicitation was performed for a range of failure modes which allowed experts to account for the range of mechanisms of failure, the potential for undiscovered mechanisms, un-inspectable areas of the containment as well as the potential for detection by alternate means. The expert elicitation process has the advantage of considering the available data for small leakage events, which have occurred in the data, and extrapolate those events and probabilities of occurrence to the potential for large magnitude leakage events.

The basic difference in the application of the ILRT interval methodology using the expert elicitation is a change in the probability of pre-existing leakage in the containment. The basic methodology uses the Jeffery's non-informative prior and the expert elicitation sensitivity study uses the results of the expert elicitation. In addition, given the relationship between leakage magnitude and probability, larger leakage that is more representative of large early release frequency, can be reflected. For the purposes of this sensitivity, the same leakage magnitudes that are used in the basic methodology (i.e., 10 L_a for small and 35 L_a for large) are used here. Table 6.2-1 illustrates the magnitudes and probabilities of a pre-existing leak in containment associated with the Jeffery's non-informative prior and the expert elicitation statistical treatments. These values are used in the ILRT interval extension for the base methodology and in this sensitivity case. Details of the expert elicitation process, and the input to expert elicitation as well as the results of the expert elicitation, are available in the various appendices of the EPRI report [22].

Table 6.2-1
EPRI Expert Elicitation Results

Leakage Size (La)	Jeffery's Non-Informative Prior	Expert Elicitation Mean Probability of Occurrence	Percent Reduction
10	2.7E-02	3.88E-03	86%
35	2.7E-03	9.86E-04	64%

A summary of the results using the expert elicitation values for probability of containment leakage is provided in Table 6.2-2. As mentioned previously, probability values are those associated with the magnitude of the leakage used in the Jeffery's non-informative prior evaluation (10L_a for small and 35L_a for large). The expert elicitation process produces a probability versus leakage magnitude relationship and it is possible to assess higher leakage magnitudes more reflective of large early releases but these evaluations are not performed in this study. Alternative leakage magnitudes could include consideration of 100 to 600 L_a where leakage begins to approach large early releases.

The net affect is that the reduction in the multipliers shown above has the same impact on the calculated increases in the LERF values. The increase in the overall probability of LERF due to Class 3b sequences that is due to increasing the ILRT test interval from 3 to 15 years is 1.43E-08/yr. Similarly, the increase due to increasing the interval from 10 to 15 years is 5.98E-09/yr. As such, if the expert elicitation mean probability of occurrences are used instead of the non-informative prior estimates, the change in LERF for Limerick is even further below the threshold criteria for a "very small" change in risk when compared to the current 1-in-10 or original 3-in-10 year requirement. The results of this sensitivity study are judged to be more indicative of the actual risk associated with the ILRT extension than the results from the assessment as dictated by the NEI methodology values, and yet are still conservative given the assumption that all of the Class 3b contribution is considered to be LERF.

Table 6.2-2

Limerick ILRT Cases: Base, 3 to 10, and 3 to 15 Yr Extensions (Based on EPRI Expert Elicitation Leakage Probabilities)

EPRI Class	DOSE Per-Rem	Base Case 3 in 10 Years		Extend to 1 in 10 Years		Extend to 1 in 15 Years	
		CDF/Yr	Per-Rem/Yr	CDF/Yr	Per-Rem/Yr	CDF/Yr	Per-Rem/Yr
1	1.32E+04	1.31E-06	1.72E-02	1.26E-06	1.67E-02	1.24E-06	1.63E-02
2	7.93E+06	2.18E-08	1.73E-01	2.18E-08	1.73E-01	2.18E-08	1.73E-01
3a	1.32E+05	1.41E-08	1.86E-03	4.69E-08	6.19E-03	7.04E-08	9.29E-03
3b	4.62E+05	3.58E-09	1.65E-03	1.19E-08	5.50E-03	1.79E-08	8.26E-03
7	3.97E+06	2.31E-06	9.15E+00	2.31E-06	9.15E+00	2.31E-06	9.15E+00
8	6.01E+06	4.65E-08	2.79E-01	4.65E-08	2.79E-01	4.65E-08	2.79E-01
Total		3.70E-06	9.62	3.70E-06	9.63	3.70E-06	9.64
ILRT Dose Rate from 3a and 3b		3.51E-03		1.17E-02		1.75E-02	
Delta Total Dose Rate	From 3 yr	---		7.63E-03		1.31E-02	
	From 10 yr	---		---		5.47E-03	
3b Frequency (LERF)		3.58E-09		1.19E-08		1.79E-08	
Delta LERF	From 3 yr	---		8.34E-09		1.43E-08	
	From 10 yr	---		---		5.98E-09	
CCFP %		64.30%		64.53%		64.69%	
Delta CCFP %	From 3 yr	---		0.23%		0.39%	
	From 10 yr	---		---		0.16%	

7.0 CONCLUSIONS

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

- 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 internal events LERF resulting from a change in the Type A ILRT test interval from three in ten years to one in fifteen years is estimated as 4.31E-8/yr using the NEI guidance as written, and at 1.43E-8/yr using the EPRI Expert Elicitation methodology. In either case, the estimated change in LERF is determined to be "very small" using the acceptance guidelines of Reg. Guide 1.174.
- The change in Type A test frequency to once-per-fifteen-years, measured as an increase to the total integrated plant risk for those accident sequences influenced by Type A testing, is 0.066 person-rem/yr using the NEI guidance, and drops to 0.013 person-rem/yr using the EPRI Expert Elicitation methodology. Therefore, in either case, the risk impact when compared to other severe accident risks is negligible.
- The increase in the conditional containment failure frequency from the three in ten year interval to one in fifteen year interval is about 1.2% using the NEI guidance, and drops to about 0.4% using the EPRI Expert Elicitation methodology. Although no official acceptance criteria exist for this risk metric, it is judged to be very small.
- Since the increase in LERF falls well below the "small change" category using the acceptance guidelines of Reg. Guide 1.174, a detailed examination of the external events impact is not required, nor would it change the conclusions from this assessment.

Therefore, increasing the ILRT interval to 15 years is considered to be insignificant since it represents a very small change to the Limerick Generating Station risk profile.

Previous Assessments

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

- Reducing the frequency of Type A tests (ILRTs) from three per 10 years to one per 20 years was found to lead to an imperceptible increase in risk. The estimated increase in risk is very small because ILRTs identify only a few potential containment leakage paths that cannot be identified by Type B and C testing, and the leaks that have been found by Type A tests have been only marginally above existing requirements.
- Given the insensitivity of risk to containment leakage rate and the small fraction of leakage paths detected solely by Type A testing, increasing the interval between integrated leakage-rate tests is possible with minimal impact on public risk. The impact of relaxing the ILRT frequency beyond one in 20 years has not been evaluated. Beyond testing the performance of containment penetrations, ILRTs also test the integrity of the containment structure.

The findings for Limerick confirm these general findings on a plant specific basis considering the severe accidents evaluated for Limerick, the Limerick containment failure modes, and the local population surrounding the Limerick Generating Station.

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