ELECTRIC POWER RESEARCH INSTITUTE

Risk Impact Assessment of Extended Integrated Leak Rate Testing Intervals

Revision 2-A of 1009325

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REPORT SUMMARY

This report presents a risk impact assessment for extending integrated leak rate test (ILRT) surveillance intervals to 15 years. The assessment demonstrates that on an industry-wide basis there is small risk associated with the extension, provided that the performance bases and defense-in-depth are maintained. There is an obvious benefit in not performing costly, critical-path, time-consuming tests that provide a limited benefit from a risk perspective.

Background
In 1995, the U.S. Nuclear Regulatory Commission (NRC) amended its regulation to provide an Option B to 10CFR50, Appendix J. Option B is a performance-based approach to leakage testing requirements in Appendix J and allows licensees with acceptable test performance history to extend surveillance intervals. At that time, provisions were made for extending ILRT frequency from three in 10 years to one in 10 years, supported by the NRC's assessment (NUREG-1493) that stated there is an imperceptible increase in risk associated with ILRT intervals up to 20 years. In about 2001, many licensees began to submit requests for one-time ILRT interval extensions of 15 years. To support changes to industry (Nuclear Energy Institute [NEI]) and the NRC guidance permitting permanent 15-year ILRT surveillance intervals) it was deemed necessary to develop the supporting risk assessment bases contained in this document.

Objectives
- To develop a generic methodology for the risk impact assessment for ILRT interval extensions to 15 years using current performance data and risk informed guidance, primarily NRC Regulatory Guide 1.174.
- To complement EPRI report, TR-104285, *Risk Impact Assessment of Revised Containment Leak Rate Testing Intervals* that considered changes to local leak rate testing and ILRT testing intervals. TR-104285 considers the change in risk based on only population dose, whereas this report considers population dose, large early release frequency (LERF) and containment conditional failure probability (CCFP).
- To exercise the methodology on two example plants and provide a template for the development of plant specific confirmatory analysis.

Approach
The first step is to obtain current containment leak rate testing performance information. The data were obtained through an NEI industry surveys, industry failure reports, and previous survey information. The data indicate that there were no failures of a magnitude that approaches that of a large release. This information is used to develop the probability of a pre-existing leak in the containment using the Jeffreys Non-Informative Prior statistical method.

A sensitivity case is performed that assesses the potential of the contribution of age-related non-inspectable containment degradation. The risk impact was then determined for two example plants, a PWR and BWR, using accident classes similar to the original EPRI report but with enhancements for assessing changes in LERF.

Results

The assessment, in conjunction with the analyses performed to date, demonstrate that there is very little risk associated with extension of ILRT intervals of 15 years. Specifically, for the conservative limiting case, the change in population dose and the change in CCFP are very small. The change in LERF for the two examples ranges from less than 10^{-7} to less than 10^{-6}, which is within the "very small" and "small" risk increase regions of Regulatory Guide 1.174. In the case where the change in LERF is greater than the very small risk increase region, the total LERF is significantly lower than the Regulatory Guide 1.174 limit for total LERF of 10^{-5} per year.

These results confirm previous conclusions in NUREG-1493 and EPRI TR-104285 regarding the low risk associated with the change in ILRT intervals using current regulatory guidance and risk-informed concepts.

EPRI Perspective

This report demonstrates that there is a small risk increase associated with the extension of ILRT intervals from the existing 10 years to 15 years. However, it is also necessary from a risk-informed perspective to maintain an awareness of and attention to defense-in-depth concepts. With respect to ILRT interval extension of 15 years, other supplemental means of verifying containment integrity, such as containment inspections, maintenance, and local leak rate testing programs, is considered necessary, as well as compliance with the ILRT performance basis requirement. Regarding the general applicability of the risk impact assessment of this document as well as insights from the 59 one-time interval extension requests referred to in Appendix G of the document, a conclusion might be reached that the small increase in risk impact is generically applicable. However, plant-specific differences may result in outliers, and it is prudent to recommend plant-specific confirmatory assessments be performed. Guidance for completing these confirmatory risk impact assessments is contained in this document.

Keywords

Integrated leak rate testing (ILRT)
Risk-informed application
Large early release frequency (LERF)
Expert elicitation
Risk impact assessment

CONTENTS

LIST OF FIGURES

LIST OF TABLES

1 INTRODUCTION

This document describes the methodology that is used to assess the risk impact associated with changes to the containment integrated leakage rate testing (ILRT) frequencies. The methodology considers the previous version of this report [1] and NUREG-1493, *Performance-Based Containment Leak-Testing Programs* [2], and builds upon the findings of these reports. In addition, submittals to the Nuclear Regulatory Commission (NRC) that proposed one-time extensions to the Type A ILRT testing interval were considered in the development of this report.

This study provides an additional analysis that supports relaxing the current Type A containment leak rate testing of 10 years to a permanent testing interval of 15 years. The additional analysis includes:

- Regulatory Guide 1.174 concepts, including the acceptable change in core damage frequency (CDF) and large early release frequency guidelines and defense-in-depth philosophy
- Sensitivity evaluations considering the impact of age-related corrosion
- Consideration of comments made on ILRT extension submittals

This document is arranged as follows:

- Section 1 – Introduction provides an introduction to the risk impact assessment of extended ILRT testing intervals.
- Section 2 – Problem Statement provides a summary of the problem statement including the more significant factors that affect the calculation of the risk impact from extended ILRT intervals.
- Section 3 – Data Applicability provides a summary of the ILRT data that have been collected and their applicability to estimating the probability of a pre-existing leak in containment.
- Section 4 – Technical Approach provides a summary of the technical approach employed in the evaluation of the risk impact associated with extended ILRT intervals.
- Section 5 – Application of the Technical Approach provides examples of the technical approach applied to two plants—one pressurized water reactor (PWR) and one boiling water reactor (BWR).
- Section 6 – Results Summary and Conclusions provides a summary of the results of the application of the technical approach to the two plants as well as a summary of the conclusions that can be drawn from these examples as well as the numerous submittals previously submitted and approved by the Nuclear Regulatory Commission (NRC).

- Section 7 – References provides a listing of the references used in the development of this document.
- Appendix A – ILRT Data provides the detailed data on containment degradations identified in the course of conducting ILRTs as well as other inspections and observations. It should be noted that the data collected were not a complete list of ILRT performed in the nuclear industry. Of particular note is that the data do not include information on the number of successful ILRTs performed without incident.
- Appendix B – Expert Elicitation Process[1] provides a summary of the process used in the expert elicitation of the relationship between the probability of a pre-existing containment leak and the magnitude of the leakage.
- Appendix C – Expert Elicitation Preparation[1] provides a summary of the methods and process used to elicit expert input.
- Appendix D – Expert Elicitation Results[1] and Analysis provides the details associated with the treatment of the expert elicited input and the final results of the pre-existing containment leakage probability versus leakage magnitude.
- Appendix E – Expert Elicitation[1] Input provides the detailed input from the experts used in the development of the pre-existing containment leakage probability versus leakage magnitude.
- Appendix F – Expert Elicitation[1] Results provides the detailed expert elicitation results.
- Appendix G – Risk Impact Assessment Submittals provides a summary of the risk impact assessment of extended ILRT intervals submittals made to the NRC.
- Appendix H – ILRT Risk Impact Assessment Template provides a suggested template for the development of plant-specific risk impact assessments of ILRT extended intervals.

[1] The expert elicitation process and results are not directly used in ILRT risk impact assessment methodology. Rather, the expert elicitation provides a portion of the basis for establishing a large leakage magnitude equal to 100 La used in the calculation of the population dose metric. The Jeffery's Non-Informative Prior method is used to determine the probability of occurrence of a large leakage event.

2
PROBLEM STATEMENT

A project was initiated to revise the industry (NEI) guidance and associated requirements for containment ILRT. Based on performance history, risk insights, and other containment testing and inspections, it is believed that the required ILRT Type A testing frequency, presently one test in 10 years, can be optimized to one test in 15 years on a permanent basis.

This project builds on the previous work documented in EPRI TR-104285, *Risk Impact Assessment of Revised Containment Leak Rate Testing Intervals* [1], and NUREG-1493, Performance-Based Leakage Test Program [2]. In fact, NUREG-1493 states, "Reducing the frequency of Type A tests (ILRTs) from the current three per 10 years to one per 20 years was found to lead to imperceptible increase in risk." Since the publication of NUREG-1493, additional containment inspections have been performed at all nuclear power plants (ASME Code Section XI Subsections IWE and IWL), and historical integrated and local leak rate testing performance has been good. Using new methods and the additional recent data, this project will demonstrate that the conclusion made in NUREG-1493 remains valid.

2.1 Background

A revision to the NEI guidance (NEI 94-01) permitting an optimized ILRT Type A testing interval of 15 years is planned. The revision will be based on a risk impact assessment that will partially supersede EPRI TR-104285, *Risk Impact Assessment of Revised Containment Leak Rate Testing Intervals* [1]. The risk impact assessment will assess the risk impact of the 15-year testing interval in a generally applicable and conservative manner and consider industry experience and appropriate regulatory guidance (RG 1.174) [4]. Two conservative (but not bounding) example plants are considered in this risk impact assessment. A template for the individual plant risk impact assessments required by NEI 94-01, Revision 2 is contained in this report as Appendix H.

Revisions to 10CFR50, Appendix J (Option B) allow individual plants to extend the ILRT Type A surveillance testing intervals on risk and performance bases. The revised Type A testing frequency is based on an acceptable performance history, defined as two consecutive periodic Type A tests at least 24 months apart in which the performance leak rate is less than the normal containment leakage of 1 La.

The basis for the current 10-year test interval is provided in Section 11 of NEI 94-01, Revision 0. NUREG-1493 contains the technical basis to support the rule-making to revise the 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 testing intervals. To supplement the NRC's rule-making basis, NEI undertook a similar study. The results of that study are documented in the Electric Power Research Institute (EPRI) research project report, TR-104285, *Risk Impact Assessment of Revised Containment Leak Rate Testing Intervals*. Both of these documents are referenced in the aforementioned NEI 94-01 basis.

The NRC report on performance-based leak testing, NUREG-1493, analyzed the effects of containment leakage on the health and safety of the public and the benefits realized from the containment leak rate testing.

The NEI Interim Guidance, promulgated in two NEI letters in November 2001 [10, 20], for performing risk impact assessments in support of ILRT interval extensions beyond 10 years builds on the EPRI Risk Assessment methodology, EPRI TR-104285. This methodology was followed in this report to determine the appropriate risk information for use in evaluating the impact of the proposed change to the ILRT testing interval.

It should be noted that containment leak-tight integrity is also verified through periodic inservice inspections conducted in accordance with the requirements of the American Society of Mechanical Engineers (ASME) Boiler and Pressure Vessel Code Section XI, Subsections IWE and IWL. These requirements will not be changed as a result of the extended ILRT interval. In addition, 10CFR50, Appendix J, Type B and C local leak tests performed to verify the leak-tight integrity of containment penetrations bellows, airlocks, seals, and gaskets are not affected by the change to the Type A test frequency.

2.2 Framework

Risk is defined as the product of probability and consequence, where probability is the periodic occurrence of an undesired event and consequence is the magnitude of the undesired event.

$$\text{RISK} = \text{PROBABILITY} \times \text{CONSEQUENCE}$$

In the case of the risk associated with the revised ILRT testing interval, the probability term in the above equation is defined as the probability of a containment leakage event that is not detected by alternative means, such as a local leak rate test or other inspection. The consequence term is defined as large early release frequency (LERF). The LERF figure of merit is one traditional figure of merit in risk-informed applications [4]. In the case of the risk impact assessment of the revised ILRT testing interval, the delta LERF is determined by multiplying the core damage frequency (CDF) by the change in the probability of a containment leakage event that would not be detected by means other than an ILRT.

The acceptance guidelines in Regulatory Guide 1.174 are used to assess the acceptability of the change in ILRT testing interval beyond that established during the Option B rule-making of Appendix J. Regulatory Guide 1.174 defines very small changes in risk as increases in CDF less than 10^{-6} per reactor year and increases in LERF less than 10^{-7} per reactor year. Since the type A test does not impact the CDF[2], the relevant risk metric is the change in LERF. Regulatory Guide 1.174 also defines small risk increase as a change in LERF less than 10^{-6} reactor year. Regulatory Guide 1.174 discusses defense-in-depth and encourages the use of risk analysis techniques to help ensure and demonstrate that key principles, such as defense-in-depth, are met.

To this end, additional figures of merit consisting of the increase in population dose (expressed both as person-rem/year and percent increase above the total base dose) and the increase in conditional containment failure probability (CCFP) (expressed as percentage points) are also developed. The population dose at the new interval is calculated by multiplying the base population dose by the change in the probability of a containment leakage event for the affected CDF end states. The metrics associated with absolute population dose change and change in population dose as % of the total dose are included in the report. Specifically, Tables 6-1 and 6-2 provide a summary of the population dose both in absolute and percentage terms. Population dose also is presented in Table 5-12 for the PWR example, Table 5-24 for the BWR example, and Table 5-8 of Appendix H for the template. CCFP is defined as the probability that the containment failed following a core damage event (for example, pre-existing containment leakage pathway). Both CCFP and the percentage point change in CCFP are presented.

Examinations of NUREG-1493 and Safety Evaluation Reports (SERs) for one-time interval extension (summarized in Appendix G) indicate a range of incremental increases in population dose[3] that have been accepted by the NRC. The range of incremental population dose increases is from ≤0.01 to 0.2 person-rem/yr and/or 0.002 to 0.46% of the total accident dose. The total doses for the spectrum of all accidents (NUREG-1493 [2], Figure 7-2) result in health effects that are at least two orders of magnitude less than the NRC Safety Goal Risk. Given these perspectives, a very small population dose is defined as an increase of ≤ 1.0 person-rem per year or ≤1 % of the total population dose, whichever is less restrictive for the risk impact assessment of the extended ILRT intervals.

Regarding CCFP, changes of up to 1.1percentage point have been accepted by the NRC for the one-time requests for extension of ILRT intervals. In context, it is noted that a CCFP of 1/10 (10%) has been approved for application to evolutionary light water designs. Given these perspectives, an increase in CCFP of ≤1.5 percentage point is assumed to be small.

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[2] In general, CDF is not significantly impacted by an extension of the ILRT interval. Plants that rely on containment overpressure for net positive suction head (NPSH) for emergency core coolant system (ECCS) injection for certain accident sequences may experience an increase in CDF. See Section 4 for further discussion of containment overpressure.

[3] The one-time extensions assumed a large leak (EPRI class 3b) magnitude of 35La, and this analysis uses 100La.

RISK	=	Probability	X	Consequence
Δ LERF	=	Δ ILRT Failure [2] Probability	X	CDF
Δ Population Dose	=	Δ ILRT Failure [4] Probability	X	Population Dose
CCFP	=	1 – (Intact CDF	/	Total CDF)

In the previous "one-time" ILRT extension submittals [3, 6], and as a matter of course in most risk-informed applications, a bounding approach was taken. This bounding approach utilized very conservative assumptions with respect to assessing the risk increase as a function of a revised ILRT testing interval. These assumptions include conservatisms associated with the determination of the ILRT failure probability as well as conservatisms associated with the determination of the consequences (delta population dose and delta LERF):

- **Data applicability.** Data used to estimate the initial probability of ILRT failure were conservatively classified. Containment leakage events that would not significantly affect population dose and/or LERF calculations were included in the estimation of the ILRT failure probability. For example, events such as steam generator manway leakage were included in the estimation of ILRT failure probability. Steam generator manway leakage would be discovered during reactor startup or during normal operation and should not impact the risk associated with an ILRT Type A testing extension.
- **No alternate means of detection.** The probability of alternate means of detection (such as local leak rate tests, inspections, or other means) was not always considered.
- **Estimation of containment leakage.** Low containment leakage rates (low La values) with higher probabilities of occurrence are used to represent a large early release.

Despite the very conservative assumptions above, the submittals to date have been able to demonstrate that the revised ILRT testing interval has little impact on risk. That is, the risk or the delta population dose and delta LERF are small.

When applying the existing methods to all plants, particularly those with higher CDF values, it is possible that some of the calculated delta LERF values will fall into the "small" change region of Regulatory Guide 1.174. In these cases a secondary test of the total LERF compared against established acceptance guidelines of 1.174 is undertaken.

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[4] The term "ILRT failure" is used in this report. The reader is reminded that in this context, "ILRT failure" is not a failure of the ILRT test to measure the containment leakage, nor does it indicate a failure of a Type A test to meet the performance criteria of NEI 94-01. Rather, the term "ILRT failure" is used to describe those ILRT tests in which containment leakage was identified above the acceptance criteria that would not be detected by a local leak rate test, containment inspections, or other alternate means and is of sufficient size to potentially result in a large early release.

2.3 Jeffreys Non-Informative Prior

The risk impact assessment of extended ILRT intervals is performed based on the Jeffreys Non-Informative Prior, as described in Sections 3 and 4. The expert elicitation process and results contained in Appendices B through F are part of the basis for establishing the large leakage rate at 100LA.

3
ILRT DATA APPLICABILITY

Data from ILRT tests have been collected at various times to support various applications. In summary, two NEI utility surveys [8, 9] and an examination of recent ILRT results provided ILRT data for 217 ILRT Type A tests that have been performed in the nuclear industry. Based on these data, the number of containment leakage events found during the performance of these tests is very small. In fact, no failures that would result in a large early release have been found. Leakage paths detectable only by Type A ILRTs and therefore influenced by the ILRT extension interval time have not been observed with magnitudes greater than 1.4 La. Consider the containment leakage or degradation event data contained in Appendix A. This Appendix A is a compilation of data from two NEI utility surveys, NUREG-1493, and other events discovered in reviewing other industry data (Licensee Event Reports [LERs], reportable events, and so on).

3.1 NUMARC Survey Data

The first ILRT survey was performed in early 1994 [8] and represented the NEI (known as NUMARC at that time) input used in NUREG-1493. In this survey, the data from 144 ILRT Type A tests were collected. Reported in NUREG-1493 were 23 ILRT failures. However, upon further review, it has been determined that these failures were conservatively classified. Of the 23 ILRT failures:

- A total of 14 were due to addition of Type B and C testing leakage penalties (local leak rate testing identified) and would not increase the time a leak path would go undetected in an ILRT interval extension.
- Four were due to steam generator in-leakage. The steam generator leak paths are identifiable during startup and normal operation and would not increase the time a leak path would go undetected in an ILRT interval extension. Leakage from the steam generators into the containment would be monitored via identified and unidentified leakage and controlled via plant technical specifications.
- Two were due to ILRT line-up errors and did not constitute valid leak paths.
- One was due to a discrepancy in a verification test and did not constitute a valid leak path.
- Two were due to failures that should have been indicated by the local leak rate testing programs. It is expected that these discrepancies would have been corrected at the next local leak rate test and therefore would not increase the time that a leak path would go undetected in an ILRT Type A interval extension.

3.2 NEI Survey Data (2001)

The second ILRT survey was performed in the fall of 2001 [9]. In the second survey, data were collected from 58 plants (91 units), reporting 38 ILRT (Type A) tests performed. The one ILRT-identified failure that should have been indicated by the local leak rate testing program would not increase the time a leak path would go undetected in an ILRT interval extension. This is because it was caused by contamination of the penetration with construction debris during a modification, which somehow passed the post-modification LLRT. However, the contamination the failure would have most likely been identified by subsequent LLRTs had the subsequent ILRT not been conducted.

3.3 Recent ILRT Data (2007)

Early in 2007, information was obtained from plants and ILRT contractors regarding recently performed ILRTs. The information extended the 2001 data to cover all but one remaining plant from the 2001 survey plus additional tests performed since 2001. Thirty-five additional Type A tests were reported, with no failures.

3.4 Combined Survey Data

In order to provide a comprehensive review of all the ILRT and applicable containment experience collected to date, the combined surveys and other data, including recent information regarding containment degradation events, were collected and are presented in Appendix A. The combined data were then sorted by those events that resulted in excessive leakage when compared with the established acceptance criteria. These include all causes that resulted in ILRT tests exceeding 1 La criteria, including those that are a result of local leak rate test penalties. A total of 71 leakage or degraded liner events are included in Appendix A. The details associated with these 71 events are provided in the appendix.

It should be noted that the combined surveys do not represent all ILRTs performed. In the initial NUMARC survey, utilities were chosen that represented a broad spectrum of reactor designs and associated ILRTs that were considered a representative sample of industry ILRTs performed. The 2001 NEI survey and the 2007 data request provided a nearly complete compilation of recent representative type A testing experience (data were obtained for 101 units). Lastly, the data collected by the surveys are supplemented by data in NUREG-1493 and additional literature searches, including LERs and reportable events.

3.5 ILRT Failure Rate Determination

From a review of the data in Appendix A and knowledge of the number of tests performed, a failure rate can be determined. In order to determine a failure rate, the number of failed events is divided by the number of demands, or in this case the number of ILRTs performed.

In order to determine the numerator (number of failed events) in the failure rate determination, a definition of what constitutes a failure must be developed. In this case, the ILRT failure is defined as the existence of a pre-existing leak in the containment that is not detected by local leak rate testing or alternate means and is detectable only in performance of an ILRT. Moreover, this pre-existing leak is capable of resulting in a LERF of fission products following a core damage accident. The definition of LERF is generally given as the exchange of a single containment volume before the effective implementation of the offsite emergency response and public protective actions [7]. In turn, public protective actions are generally assumed to be taken approximately 2 to 4 hours following a core damage event. The exchange of a single containment volume within a 4-hour period corresponds to a leakage rate of 600% per day or 600 to 6000 La (assuming that the ILRT acceptance criterion for the plant in question is between 1% and 0.1% per day).

Some previous submittals have conservatively assumed (based on Reference [1]) that four failures have occurred (based on the 1994 NUMARC survey). However, based on a more comprehensive review of the data, no containment leakage events with leakage greater than 21 La have been discovered[5]. As discussed further in Section 3.6, events with leakages from 600 La to 6000 La are a more realistic representation of a large early release. Previous submittals (specifically, Reference [3]) conservatively assumed that events with a leakage greater than 35 La were capable of producing a large early release.

Using any definition of large early release greater than the minimum 35 La (from Reference [3]), there are no containment leakage events that could result in a large early release in the current dataset. The zero failures are based on the combined ILRT database (NUMARC and NEI surveys [8, 9] and other sources), in which the results of 217 ILRTs have been documented. (It is conservatively estimated that over 400 ILRTs have been performed in the U.S. nuclear industry. The 217 ILRTs that have been documented are used in this submittal and analysis.)

With zero failed events, a variety of statistical methods is available to estimate a failure rate. However, for the purposes of this analysis, it is believed that the Jeffreys Non-Informed Prior provides a reasonable balance between conservatism in light of uncertainty and yet meets the intent of RG 1.174.

$$\text{Jeffreys Failure Probability} = \frac{\text{Number of Failures} + 1/2}{\text{Number of Tests} + 1}$$

For no failures and 217 tests, the probability = 0.5/217 = 0.0023

[5] There are several tests where the resulting leakage was indicated as above the acceptance criteria but not quantified. The reasons for not quantifying leakage are not clear but could include leakage exceeding instrument ranges or a desire to simply correct the path without quantifying the as-found data. Based on available information, the magnitude of these leak paths is not expected to exceed that of known, quantified leak paths.

3.6 No Alternate Means of Detection

Various alternative methods of detecting a leakage pathway (“ILRT failure”) in containment exist. These methods include local leak rate tests (LLRTs), reactor startup, normal operation, and other containment and piping inspections. Since the publication of NUREG-1493, additional containment inspections have been performed at all nuclear plants (ASME Code Section XI, Subsections IWE and IWL).

In addition, experience has shown that during normal reactor startup and during normal power operation, it is fairly routine for most containment designs to either vent the overpressure that has built up or to provide nitrogen makeup (for inerted containment designs) to maintain positive pressure within specified limits. The increase in pressure can be caused by increase in the average air temperature during heatup and startup, changes in barometric pressure, and an increase in the containment air mass from compressed air equipment bleeds and leakage. Absence or significant changes in the frequency of pressure buildup, and venting over a substantial period of time will provide a qualitative indication of the existence of a containment atmosphere to outside atmosphere leak path. These factors, as well as others, provide additional means of detection of containment leakage pathways.

3.7 Estimation of Containment Leakage

Previous one-time ILRT extension submittals have used an estimated leakage rate as a result of an assumed large ILRT failure of 35 La [3, 6, 10]. This leakage was assumed to conservatively represent the leakage rate associated with a large early release as calculated in the Level 2 probabilistic risk assessment (PRA). However, the definition of LERF is generally given as the exchange of a single containment volume before the effective implementation of the offsite emergency response and public protective actions [7]. In turn, public protective actions are generally assumed to be taken approximately 2 to 4 hours following a core damage event. The exchange of a single containment volume within a 4-hour period corresponds to a leakage rate of 600% per day or 600 to 6000 times La, assuming that the ILRT acceptance criteria for the plant in question are between 1% and 0.1% per day. While the leakage value 35 La used in the one-time ILRT interval extension requests is very conservative, 100 La is used in this analysis to represent leakage magnitudes capable of producing a large early release. The expert elicitation in Appendices B-F of this report has not been fully accepted by the NRC. However, the range of large leakages in the expert elicitation indicates that an estimated leakage of approximately 100 La is the frequency weighted average of a reasonable range of leakage magnitudes.

From an examination of the events with stated leak rates in Appendix A, the highest known leakage event has a leakage of 21 La (event number 10). This event was discovered during performance of an LLRT. The next highest leakage event has a leakage of 15 La (number 35) and was discovered during the performance of the ILRT. However, this event was the result of excessive local leakage that would be discovered during the next LLRT.

Therefore, there are no events that have occurred in the database that would constitute a large early release pathway. In fact, the use of 100 La to represent a large early release is very conservative given the definition provided in this evaluation.

However, the data collected do provide useful information on the type of failures that have occurred, the potential failure mechanisms, and the historical sizes of these failures. Various sorts were performed on the data to better understand the available information and the conclusion that can be drawn from them.

Of the 71 events in the ILRT database, 32 involved leakages ≤ 1 La; the remaining 39 events have unknown leakages or leakages greater than 1 La. Of these 39 events, 20 were identified by local leak rate testing (18) or involved steam generator manway leakage (2). Because steam generator manway leakage will result in a loss of steam generator water (secondary side) to the containment during reactor startup and normal operation and identified and unidentified leakage is monitored in technical specifications, these can be removed from consideration.

Of the remaining 19 events, three are the result of the previous practice of performing an ILRT prior to completing local leak rate tests. This results in the ILRT discovering leakages that would normally be found during a local leak rate test. These events are indicated in Appendix A with the phrase “ILRT prior to LLRT” in the description column.

Of the 16 remaining events, seven were discovered by alternate means (not impacted by extension of ILRT intervals), specifically by operator or other inspections. It is assumed for these seven events that the frequency of detection and ILRT failure frequency would remain constant regardless of testing because no changes to the frequency of other tests or inspections are proposed. Therefore, these seven events are not considered in the calculation of the ILRT failure rate. In addition, one event is the result of instrumentation problems and does not appear to be an actual ILRT failure.

The nine remaining events are presented below. The sizes in terms of leakage rates of the nine events are as follow:

- Unknown leakage events: 4
- Small leakage events (<2 La): 3
- Medium leakage events (2–10 La): 1
- Large leakage events (>10 La): 1

Of these nine events, three events (Nos. 34, 35, and 61) represent LLRT failures to discover leakage, and one event (No. 41) represents failure of the drywell head seal due to relaxation of improper spherical washer material. In the case of the LLRT failures that should have been identified by local leak rate testing, the leakage would most likely be detected during the performance of the next LLRT and therefore does not affect the ILRT failure rate for the purposes of ILRT testing interval extension. In the case of the drywell head leakage, this event would be identified and corrected in the next refueling outage and therefore does not impact the ILRT failure frequency with regard to ILRT testing interval extension.

The remaining five events were detected by the ILRT. In four of the five events, the estimated leakage is unknown. The fifth event (No. 45) falls into the small leakage category (1.4 La). Of these five events, two events could have been detected only by conducting an ILRT (Nos. 1 and

45). However, these events had either unknown leakage rates or leakage rates less than 2 La. One event (No. 1) involved two holes drilled in a liner (unknown leakage rate), and the other (No. 45) involved the ejection of a radiation monitor during an ILRT (1.4 La).

Event 30 is of unknown leakage and unknown cause. Two events (Nos. 25 and 33) should have been detected by an LLRT and were not. NEI 94-01 does not allow extension of the ILRT interval if the performance criteria cannot be met. That is, if a leak path involving a penetration cannot be determined by an LLRT, the ILRT interval cannot be extended (NEI 94-01, Section 9.1.1).

In summary, from a detailed review of the available data, there have been no events identified that could have resulted in a large early release as currently defined. Several ILRT events had unknown leakage rates. From the description of the events it can be inferred, although not proven, that the leakage was not large (for example, holes drilled in liner and penetration leakage). In any event, the limited ILRT data result in an inability to directly calculate an ILRT failure rate. However, the information that the data provide is valuable in an expert elicitation designed to estimate the probability of ILRT failure rates for a wide magnitude of leakage rates.

4 TECHNICAL APPROACH

The guidance provided in this report section builds on the EPRI Risk Impact Assessment Methodology [1] and the NRC Performance-Based Containment Leakage Test Program [2] and is consistent with applicable risk-informed decision-making principles of NRC Regulatory Guide 1.174 [4]. This assessment methodology also considers approaches utilized in various utility submittals, including Indian Point 3 (and the associated NRC SER) and Crystal River [3, 5, 6].

It was not considered possible for this report to anticipate or cover all possible risk-informed applications that may result in changes to containment testing or maintenance. For example, some BWR plants have applied a risk-informed method to support extension of a drywell bypass test interval. Other risk-informed applications might include plant-specific exemptions from regulatory requirements for testing or maintenance. Such changes, if any, and as applicable, should be accounted for in the plant-specific assessment of the risk associated with the extension of the ILRT interval.

4.1 Methodology Improvements

The guidance in this report section improves on the above methods in four areas, specifically, improved calculation of risk increase, ILRT failure frequency and magnitude, improved estimation of population dose, and consideration of the potential for liner corrosion.

The first area involves the methodology for determining the impact resulting from extending surveillance intervals. References [1] and [2] both consider the percentage increase in the probability of leakage as an appropriate multiplier to be used in risk impact dose calculations. It is now believed that the multiplier used should be a factor representing the change in probability of leakage. As stated in References [1] and [2], relaxing the test frequency from three in 10 years to one in 10 years increases the average time that a leak detectable only by an ILRT would go undetected from 18 (three years/2) to 60 (10 years/2) months. This is a factor of 60/18 = 3.333.

The baseline dose determined in the EPRI report was $7x10^{-3}$ person-rem per year, and the dose associated with the 10-year interval was calculated using the percentage increase (10%), or 1.1 times the baseline, $7.7x10^{-3}$ person-rem per year. However, using the revised assessment cited above and the resulting factor of 3.33 would yield a 10-year dose of 3.33 x 7x10-3 = 2.3x10-2 person-rem per year[6]. The 10-year dose increase is still a very small risk contribution,

[6] The EPRI report was based on the logic that because ILRTs detect only 3% of leaks, the factor of a 3.333 increase results in a change in the overall probability of leakage from 3% to 3 * 3.333 = 10%, or a 10% increase in the baseline dose. The baseline dose determined in the EPRI report was $7x10^{-3}$ person-rem/yr, and the dose associated with the 10-year interval was calculated as a 10% increase or 1.1 times the baseline, $7.7x10^{-3}$ person-rem/yr. It is

only 0.11% of the total dose of 22 person-rem per year. This represents an increase in risk of 0.078% from the baseline contribution of 0.032%. The small increase in total dose results because ILRTs address a very small portion of the severe accident risk. NUREG-1493 reported a similar 0.07% risk increase for Surry under the same assumptions and interval extension.

The second improvement area is in the methodology used to determine the frequencies of leakages detectable only by ILRTs, Classes 3a and 3b. The method utilized in the aforementioned utility submittals involved using a 95% confidence of the distribution of the noted ILRT failures (four of 144 reported in NUREG-1493). Data collected recently by NEI from 101 nuclear power plants indicate that 73 plants have conducted ILRTs since January 1995, with only one failure (due to construction debris from a penetration modification). The information presented at the end of Section 3.7 clearly indicates that only two events could have been detected only during the performance of an ILRT and thus impact risk due to a change in ILRT frequency. This consideration and recent information regarding the number of successful ILRTs (217) would indicate that the statistical information should be based on two failures out of 217 tests. Rather than using the 95% confidence of the distribution, it is considered more appropriate (and more conservative) to utilize the maximum likelihood estimate (arithmetic average) (2/217 = 0.0.0092) for the class 3a distribution and Jeffreys Non-Informative Prior distribution [7] for the class 3b distribution.

The third improvement includes provisions for utilizing representative plant dose calculations that are related to NUREG-1150 doses[7]. This approach will be employed in this report for the industry-wide generic assessments conducted to assess the risk impact of optimized extension of ILRT intervals. However, if an individual plant desired to conduct a plant-specific assessment and the plant information was available in the plant PRA, it could be utilized.

The fourth improvement involves the treatment of the potential for liner corrosion. In the Calvert Cliffs Response to the Request for Additional Information Concerning the License Amendment for a One-Time Integrated Leak Rate Test Extension [17], a method for determining the change in likelihood of detecting liner corrosion and corresponding change in risk due to the ILRT extension is provided. This method is applied in this generic submittal of the risk impact of the ILRT extension.

4.2 Methodology Steps

The EPRI methodology [1] employed a simplified risk model utilizing a PRA containment event tree (CET), which provides a risk framework for evaluating the effect of containment isolation

now believed that the dose associated with the 10-year interval should have been calculated based on the change in the probability of leakage, 3.333, rather than the factor of 1.10. The argument above shows this difference in test interval effect on leakage probability to not affect the overall conclusions with regards to population dose as a function of ILRT interval changes.

[7] EPRI report TR-104285 developed consequence measures in terms of population dose for each accident class. The analysis required defining offsite consequences. While the representative plants were not NUREG-1150 plants, this analysis used the MACCS consequence (population dose) calculations conducted for NUREG-1150, Surry, and Peach Bottom. See page 4-5 of the EPRI report for more detail.

failures affected by leakage testing requirements. The complexity of the CET models, however, is not necessary to evaluate the impact of containment isolation system failures. Therefore, a simplified risk model was developed to distinguish between those accident sequences that are affected by the status of the containment isolation system versus those that are a direct function of severe accident phenomena. The simplified risk model allowed for a smaller number of CET scenarios to be evaluated to determine the baseline risk as well as subsequent analysis to quantify risk effects of extending test intervals. The methodology regrouped core damage accident sequences reported in PRAs that were reviewed in the study into eight classifications to permit the aforementioned differentiation. See Table 7-1 for a description of the eight end-state classifications. The risk metric was defined as the product of frequency and consequence (person-rem/reactor-year).

The Indian Point Methodology [3] quantifies leakage from accident sequences in end states (3a and 3b). Accident sequence end states 3a and 3b have the potential to result in a change in risk associated with changes in ILRT intervals because a pre-existing leak is assumed to be present for these end states. By manipulating the probability of a pre-existing leak of sufficient leak size, an evaluation of the change in LERF can be performed. The NRC [5] considered this an improvement on the EPRI study. Similar information is contained in the Crystal River submittal [6].

This assessment guidance incorporates these and other features of the above methodologies. The first three 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 fourth step in the methodology calculates the change in LERF and compares it to the guidelines in Regulatory Guide 1.174. Because there is no change in CDF [1], the change in LERF suffices as the quantitative basis for a risk-informed decision per current NRC practice, namely Regulatory Guide 1.174. The fourth step also calculates the change in containment failure probability. The NRC has previously accepted similar calculations [2], referred to as conditional containment failure probability (CCFP), as the basis for showing that the proposed change is consistent with the defense-in-depth philosophy. As such, this step suffices as the remaining basis for a risk-informed decision per Regulatory Guide 1.174. The fifth and final step assesses the impact of extended intervals on containment liner corrosion in a sensitivity analysis. A summary of the steps is as follows:

1. Quantify the baseline (three-year ILRT frequency) risk in terms of frequency per reactor year for the EPRI accident classes of interest.

2. Develop the baseline population dose (person-rem, from the plant PRA or IPE, or calculated based on leakage) for the applicable accident classes.

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

4. Determine the risk impact in terms of the change in LERF and the change in CCFP.

5. Consider both internal and external events.

6. Evaluate the sensitivity of the results to assumptions in the liner corrosion analysis.

The methodology is employed to assess the risk impact of extending optimized ILRT intervals to 15 years. Representative plant assessments are provided in Section 5.

4.2.1 Step One: Baseline Risk Determination

In this step, the baseline risk is determined in terms of core damage frequency per reactor year for the EPRI accident classes[8], excluding accident classes 4, 5, and 6. EPRI accident classes 4, 5, and 6 are excluded because ILRT Type B and C tests and multiple failures of redundant isolation valves to stroke closed are not impacted by changes in ILRT frequency, and their contribution to population dose is small. The determination of the baseline risk is accomplished as follows:

1. Referring to the plant PRA or IPE, obtain core damage frequency (CDF) values for the EPRI accident classes 1, 2, 7, and 8 or the plant-specific accident class equivalent. This may require reclassification of the PRA sequences into the EPRI bins, which can be accomplished using the definitions of the EPRI accident classes contained in Table 4-1.
2. Determine the frequencies for Class 3a and Class 3b as follows:
   - frequency = CDF * Class 3a leakage probability
   - frequency = CDF * Class 3b leakage probability

   To calculate the probability that a liner (or other leak path not monitored by local leak rate testing and/or alternate means) leak will be large (accident Class 3b), the Jeffreys Non-Informative Prior is used as the Class 3b Leakage Probability.
   A similar approach is used to calculate the probability that a liner leak will be small (accident Class 3a), using available data and the Class 3a Leakage Probability. In the case of the small pre-existing leakage, the probability is taken from reference 3 and is consistent with previous risk impact assessments of extended ILRT interval submittals.
3. Adjust the Accident Class 1 frequency as (individual plant examination [IPE] Class 1) minus (Class 3a and Class 3b). This is necessary to maintain the sum of the frequencies of the accident classes equal to the CDF.
4. NEI supplemental guidance [20] to the NEI Interim Guidance [10] provides additional information concerning the conservatisms in the quantitative calculation of delta LERF. The supplemental guidance describes methods, using plant-specific calculations to address the conservatisms. The supplemental guidance 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 the 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."

[8] See Section 4.3 for a complete description of the EPRI accident classes.

- An example of the type of sequences that may independently cause LERF is a sequence associated with containment bypass events, such as steam generator tube rupture (SGTR) or interfacing system loss of coolant accidents (ISLOCA). Another example may include those accident sequences associated with anticipated transient without scram (ATWS) events.
- An example of the type of sequence that may never result in LERF is a sequence where containment sprays and containment heat removal are available. In these sequences, containment sprays and cooling reduce the fission products via scrubbing and rapidly reduce containment pressure. The basis for the removal of sequences to reduce conservatism is plant- and PRA-specific and should be documented by analysis in the risk impact assessment.

4.2.2 Step Two: Develop the Baseline Population Dose

In this step, the baseline population dose (person-rem, from the plant PRA, or calculated based on leakage) is developed for the applicable accident classes. The most relevant plant-specific information should be used to develop population dose information. The order of preference shall be plant-specific best estimate, Severe Accident Mitigation Alternative (SAMA) for license renewal, and scaling of a reference plant population dose.

1. From the plant IPE or PRA, determine the relationship between offsite dose (person-rem) and containment leakage rate (the dose in person-rem) for Class 1, which is assumed to be equal to 1.0 La.
2. From the plant IPE, determine the offsite dose (person-rem) for the accident classes where analysis is available, typically Classes 1, 2, 7, and 8.
3. For those accident classes where analysis is not available in the IPE or PRA, determine the dose by first determining the class containment leak rate and multiplying by the 1.0 La dose.
4. For accident Classes 3a and 3b leak rates, conservative values of 10 La and 100 La, respectively, are used.
5. Determine the baseline accident class dose rates (person-rem/year) by multiplying the dose by the frequency for each of the accident classes. Sum the accident class dose rates to obtain the total dose rate.

6. For the cases where plant-specific PRA dose information is not available, a representative population dose can be calculated using other references, such as NUREG/CR-4551. To develop a representative population dose, the NUREG/CR-4551 plant that most closely resembles the analysis plant is chosen. Relate the NUREG/CR-4551 accident progression bins (APBs), EPRI Accident Classes, and plant-specific plant damage states (PDSs) based on the definitions contained in NUREG/CR-4551, Table 4-1 and plant-specific PDS. Adjust the resulting EPRI Accident Class 1, 2, 7, and 8 population doses to account for substantial differences in reactor power level, population density, allowable containment leak rate (La), and other plant-specific factors that may affect population dose as follows:
   - Population density adjustment=
     (population within 50 miles of the plant ÷ population within 50 miles of the reference plant, from reference document)
   - Power level adjustment=
     Rated power level of plant (MWt) ÷ rated power level of reference plant
   - La adjustment=
     La of plant (%w/o/day) ÷ La of reference plant
7. The population density and power level adjustments are applicable to all EPRI accident classes; however, the La adjustment should be made only to intact containment end states.
8. For EPRI Accident Classes 3a and 3b, determine the population dose by multiplying the population dose of Accident Class 1 by 10 La and 100 La, respectively.

4.2.3 Step Three: Evaluate the Risk Impact (Bin Frequency and Population Dose)

In this step, the risk impact associated with the change in ILRT testing intervals is evaluated.

1. Determine the change in probability of leakage detectable only by ILRT (Classes 3a and 3b) for the new surveillance intervals of interest. NUREG 1493 [5] states that relaxing the ILRT frequency from three in 10 years to one in 10 years will increase the average time that a leak that is detectable only by ILRT goes undetected from 18 to 60 months (1/2 the surveillance interval), a factor of 60/18 = 3.33 increase. Therefore, relaxing the ILRT testing frequency from three in 10 years to one in 15 years will increase the average time that a leak that is detectable only by ILRT goes undetected from 18 to 90 months (1/2 the surveillance interval), a factor of 90/18 = 5.0 increase.
2. Determine the population dose rate for the new surveillance intervals of interest by multiplying the dose by the frequency for each of the accident classes. Sum the accident class dose rates to obtain the total dose rate.
3. Determine the increase in dose rate and percentile increase for each extended interval as follows: Increase in dose rate = (total dose rate of new interval minus total baseline dose), and percent increase = [(increase in dose rate) divided by (total baseline dose rate)] x 100.

4.2.4 Step Four: Evaluate Change in LERF and CCFP

In this step, the changes in LERF and CCFP are evaluated.

1. Evaluate the risk impact in terms of change in LERF. The risk associated with extending the ILRT interval involves a potential that a core damage event that normally would result in only a small radioactive release from containment could result in a large release due to an undetected leak path existing during the extended interval. As discussed in References [1] and [2], only Class 3 sequences have the potential to result in early releases if a pre-existing leak were present. Late releases are excluded regardless of the size of the leak because late releases are not, by definition, LERF events. The frequency of class 3b sequences is used as a measure of LERF, and the change in LERF is determined by the change in class 3b frequency. Refer to Regulatory Guide 1.174 [4] for LERF acceptance guidelines.

   ΔLERF = (frequency class 3b interval x) - (frequency class 3b baseline).

2. Evaluate the change in CCFP. The conditional containment failure probability is defined as the probability of containment failure given the occurrence of a core damage accident, which can be expressed as:

   CCFP = [1 - (frequency that results in no containment failure)/CDF] * 100%

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

   CCFP Change (increase) = (CCFP at interval x) – (CCFP at baseline interval), expressed as percentage point change.

4.2.5 Step Five: Evaluate Sensitivity of Results

In this step, the risk impact results sensitivity to assumptions in liner corrosion are investigated.

Evaluate the sensitivity of the impact of extended intervals to liner corrosion. The methodology developed for Calvert Cliffs investigates how an age-related degradation mechanism can be factored into the risk impact associated with longer ILRT testing intervals.

> Note: NRC Information Notice 2004-09 [22] describing the Brunswick and other events and voluntary LER 1999-002-00 for North Anna Power Station Unit 2 [23] provide information on corrosion of steel containment and containment liner. The instances of through-wall penetration described in these documents are considered in the development of the risk assessment methodology and are part of the plant-specific analyses performed for assessing the potential for liner corrosion.

4.2.6 Containment Overpressure

In general, CDF is not significantly impacted by an extension of the ILRT interval. Plants that rely on containment overpressure for net positive suction head (NPSH) for emergency core coolant system (ECCS) injection for certain accident sequences may experience an increase in CDF.

In the case of accident sequences that are the result of the long-term loss of containment heat removal, containment pressurization and eventual failure are assumed to result in a loss of core coolant injection systems.

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

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

Either of these scenarios could be impacted by a large containment failure that eliminates the overpressure contribution to the available NPSH calculation. If either of these cases is susceptible to whether or not containment overpressure is available (or other cases are identified), then the PRA model should be adjusted to account for this requirement. As a first-order estimate of the impact, it can be assumed that the EPRI Class 3b contribution would lead to loss of containment overpressure, and the systems that require this contribution to NPSH should be made unavailable when such an isolation failure exists. The impact on CDF can then be accounted for in a similar fashion to the LERF contribution as the EPRI Class 3b contribution changes for various ILRT test intervals. The combined impacts on CDF and LERF should then be considered in the ILRT evaluation and compared with the Regulatory Guide 1.174 acceptance guidelines. These are recommended considerations for plants that take credit for containment overpressure in ECCS recirculation analyses, however the level of detail in this report is not sufficient for a proper assessment of the impacts of this situation. As is indicated in Section H.1.2 of this report, a traditional license amendment request (including the risk impact assessment) may be required for those cases where containment overpressure is credited in the ECCS analysis.

4.2.7 External Events

Where possible, the analysis should include a quantitative assessment of the contribution of external events (for example, fire and seismic) in the risk impact assessment for extended ILRT intervals. For example, where a licensee possesses a quantitative fire analysis and that analysis is of sufficient quality and detail to assess the impact, the methods used to obtain the impact from internal events should be applied for the external event. If the external event analysis is not of sufficient quality or detail to allow direct application of the methodology provided in this document, the quality or detail will be increased or a suitable estimate of the risk impact from the external events should be performed. This assessment can be taken from existing, previously submitted and approved analyses or another alternate method of assessing an order of magnitude estimate for contribution of the external event to the impact of the changed interval.

Section 5, application of the technical approach, provides an example of the assessment of external event LERF.

4.3 EPRI Accident Class Descriptions

Extension of the Type A interval does not influence those accident progressions that involve containment isolation failures associated with Type B or Type C testing or containment failure induced by severe accident phenomena. The CET containment isolation models are reviewed for applicable isolation failures and their impacts on the overall plant risk. Specifically, a simplified model to predict the likelihood of having a small or large pre-existing breach in the containment that is undetected due to the extension of the Type A ILRT test interval is developed. For this work, the EPRI accident classes are used to define the spectrum of plant releases. The intact containment event was modified to include the probability of a pre-existing containment breach at the time of core damage. Two additional basic events are addressed. These are Event Class 3a (small leak) and Class 3b (large leak). (This addresses the "Class 3" sequence discussed in EPRI TR-104285.) Both event Class 3a and 3b are considered in estimating the public exposure impact of the ILRT extension. However, since leaks associated with event Class 3a are small (that is, marginally above normal containment leakage), only event Class 3b frequency change is considered in bounding the LERF impact for the proposed change. The eight EPRI accident classes are discussed in the following paragraphs.

Class 1 sequences: This sequence class consists of all core damage accident progression bins for which the containment remains intact with negligible leakage. Class 1 sequences arise from those core damage sequences where containment isolation is successful and long-term containment heat removal capability is available. The frequency of an intact containment is established based on the individual plant's PRA. For Class 1 sequences, it is assumed that the intact containment end state is subject to a containment leakage rate less than the containment allowable leakage (La). To obtain the Class 1 event frequency, intact containment events are parsed into three classes: Class 3a, Class 3b, and Class 1. Class 1 represents containments with expected leakages less than La. Class 3a represents intact containments with leakages somewhat larger than La, and Class 3b represents intact containment end states with large leaks. The frequency for Class 1 events is related to the intact containment core damage frequency (CDF_{Intact}) and the Class 3 categories, as follows.

$$F_{Class\ 1} = CDF_{Intact} - F_{Class\ 3a} - F_{Class\ 3b}$$

Where:

CDF_{Intact} = the core damage frequency for intact containment sequences from the plant-specific PRA.

The calculation of Class 3 frequencies is discussed below. Radiological releases for Class 1 sequences are established assuming a containment leakage rate equal to the design basis allowable leakage (La).

Class 2 sequences: This group consists of all core damage accident progression bins for which a pre-existing leakage due to failure to isolate the containment occurs. These sequences are dominated by failure to close of large (>2 inches [5.1 cm] in diameter) containment isolation valves. The frequency per year for these sequences is determined from the plant-specific PRA as follows:

$$F_{Class\ 2} = PROB_{large\ CI} * CDF_{Total}$$

Where:

$PROB_{large\ CI}$ = random containment large isolation failure probability (large valves), and

CDF_{Total} = total plant-specific core damage frequency, which is obtained from plant-specific PRA.

Class 3 sequences: Class 3 end states are developed specifically for this application. The Class 3 end states include all core damage accident progression bins with a pre-existing leakage in the containment structure in excess of normal leakage. The containment leakage for these sequences can be grouped into two categories: small leakage or large. The respective frequencies per year are determined as follows:

$$F_{Class\ 3a} = PROB_{Class\ 3a} * CDF$$

$$F_{Class\ 3b} = PROB_{Class\ 3b} * CDF$$

Where:

$PROB_{Class\ 3a}$ = the probability of small pre-existing containment leakage in excess of design allowable but less than 10 La. $PROB_{Class\ 3a}$ is a function of ILRT test interval.

$PROB_{Class\ 3b}$ = the probability of large (100 La) pre-existing containment leakage. $PROB_{Class\ 3b}$ is a function of ILRT test interval.

CDF_{Total} = total plant-specific core damage frequency, which is obtained from plant-specific PRA.

While no pre-existing leakage in excess of 21 La has been identified for any historical ILRT event, Class 3b releases are conservatively assessed at 100 La. Class 3a releases are conservatively assessed at 10 La.

Class 4 sequences: This group consists of all core damage accident progression bins for which a failure-to-seal containment isolation failure of Type B test components occurs. Because these failures are detected by Type B tests and their frequency is very low compared with the other classes, this group is not evaluated any further. The frequency for Class 4 sequences is subsumed into Class 7, where it contributes insignificantly.

Class 5 sequences: This group consists of all core damage accident progression bins for which a failure-to-seal containment isolation failure of Type C test components occurs. Because these failures are detected by Type C tests and their frequency is very low compared with the other classes, this group is not evaluated any further. The frequency for Class 5 sequences is subsumed into Class 7, where it contributes insignificantly.

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, typically resulting in a failure to close smaller containment isolation valves. All other failure modes are bounded by the Class 2 assumptions. This accident class is also not evaluated further.

Class 7 sequences: This group consists of all core damage accident progression bins in which containment failure induced by severe accident phenomena occurs (for example, H_2 combustion and direct containment heating):

$$F_{\text{Class 7}} = CDF_{CFL} + CDF_{CFE}$$

Where:

CDF_{CFE} = the core damage frequency resulting from accident sequences that lead to early containment failure, and

CDF_{CFL} = the core damage frequency resulting from accident sequences that lead to late containment failure.

$F_{\text{Class 7}}$ can be determined by subtracting the intact, bypass (see Class 8 discussion) and loss of isolation CDFs from the total CDF. These end states include containment failure.

Class 8 Sequences: This group consists of all core damage accident progression bins in which containment bypass occurs. Each plant’s PRA is used to determine the containment bypass contribution. Contributors to bypass events include ISLOCA events and SGTRs with an unisolated steam generator.

$$F_{\text{Class 8}} = CDF_{ISLOCA} + CDF_{\text{Unisolated SGTR}}$$

The magnitude of bypass releases is plant-specific and is typically considerably larger (two or more orders of magnitude) than releases expected for leakage events. The containment structure will not impact the release magnitude for this event class.

A complete list and description of the EPRI accident classes are displayed in Table 4-1.

Table 4-1
Description of the EPRI Accident Classes

Class No.	Description	Frequency	Leakage	Population Dose (person-rem)	Population Dose Rate (person-rem/rx-yr)
1	Containment intact	Calculated value $F_{Class\ 1} = CDF_{Intact} - F_{Class\ 3a} - F_{Class\ 3a}$	La	Value from plant PRA, or EPRI / NUREG 1150, or NUREG/CR-4551	Dose 1 * Frequency 1
2	Large containment isolation failures	Value from plant PRA $F_{Class\ 2} = PROB_{large\ CI} * CDF_{Total}$	Value from plant PRA	Value from plant PRA, or EPRI / NUREG 1150, or NUREG/CR-4551	Dose 2 * Frequency 2
3a	Small pre-existing leak in containment	Calculated value $F_{Class\ 3a} = PROB_{Class\ 3a} * CDF$	10 La	(Class 1 dose for La) * 10 La	Dose 3a * Frequency 3a
3b	Large pre-existing leak in containment	Calculated value $F_{Class\ 3b} = PROB_{Class\ 3b} * CDF$	100 La	(Class 1 dose for La) * 100 La	Dose 3b * Frequency 3b
4	Small isolation failure – failure to seal – (Type B test)	NA	NA	NA	NA
5	Small isolation failure – failure to seal - (Type C test)	NA	NA	NA	NA
6	Containment isolation failures (dependent failures personnel errors)	NA	NA	NA	NA
7	Severe accident phenomena-induced failures (early and late containment failures)	Value from plant PRA $F_{Class\ 7} = CDF_{CFL} + CDF_{CFE}$	Value from plant PRA	Value from plant PRA, or EPRI / NUREG 1150, or NUREG/CR-4551	Dose 7 * Frequency 7
8	Containment bypass (SGTR, MSIV leakage, and ISLOCA)	Value from plant PRA $F_{Class\ 8} = CDF_{ISLOCA} + CDF_{Uniso\ SGTR}$	Value from plant PRA	Value from plant PRA, or EPRI / NUREG 1150, or NUREG/CR-4551	Dose 8 * Frequency 8

CDF_{Intact} = the core damage frequency for intact containment sequences from the plant-specific PRAs

$PROB_{large\ CI}$ = random containment large isolation failure probability (i.e. large valves)

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

$PROB_{Class\ 3a}$ = the probability of small (10 La) pre-existing containment leakage

$PROB_{Class\ 3b}$ = the probability of large (100 La) pre-existing containment leakage

CDF_{CFE} = the core damage frequency resulting from accident sequences that lead to early containment failure

CDF_{CFL} = the core damage frequency resulting from accident sequences that lead to late containment failure

5
APPLICATION OF TECHNICAL APPROACH

In this report section, the technical approach outlined in Section 4 is applied to two plants for the purpose of illustrating the application of the methodology. The first plant is a PWR with a large dry containment. The second plant is a BWR with a small containment (that is, drywell-torus combination). The data for both plants are based on actual plant-specific data, and both plants have made ILRT Test Interval Extension submittals to the NRC that have been approved. The PWR plant is based on the Vogtle Electric Generating Station [18], and the BWR plant is based on the Columbia Generating Station [19].

The five-step process outlined in Section 4 of this report is applied. The individual report subsections below correspond to the steps outlined in the process.

5.1 PWR Example

This example provides the details of the methodology applied to Vogtle Electric Generating Plant (VEGP) operated by Southern Nuclear Operating Company (SNC). Large portions of this example are adapted from the Vogtle submittal [18].

It should be noted that the Vogtle PRA has a large core damage frequency contribution to the intact end state (that is, Class 1). While most PWR PRAs generally have a large contribution to the intact end state or Class 1, the magnitude of the Vogtle contribution is considered high. This high intact CDF is conservative from the perspective of the change in population dose, LERF, and CCFP. Therefore, while not typical, the Vogtle case is generally bounding of most other PWR plants from these perspectives. Plant-specific features and PRA results should be reflected in the plant-specific analysis performed.

The VEGP level 2 model was developed to calculate the large early release frequency (LERF) as well as the other release categories. The total LERF, 5.89E-08 per year, corresponds to VEGP release categories D, G, and T in Table 5-1.

Table 5-1
VEGP Release Category Frequency

Release Category	VEPG Release Category Definition	Frequency (per year)
A	No containment failure within 48-hour mission time, but failure could eventually occur without further mitigating action; noble gases and less than 0.1% volatiles released	1.42E-05
D	Containment bypassed with noble gases and up to 10% of the volatiles released	4.26E-09
G	Containment failure prior to vessel failure with noble gases and up to 10% volatiles released (containment not isolated)	5.98E-10
K	Late containment failure with noble gases and less than 0.1% volatiles released (containment failure greater than 6 hours after vessel failure; containment not bypassed; isolation successful prior to core damage)	2.23E-08
S	Success (leakage only, success maintenance of containment integrity; containment not bypassed; isolation successful prior to core damage)	4.32E-09
T	Containment bypassed with noble gases and more than 10% of the volatiles released	5.40E-08
	Total Release Category Frequency	1.42E-05
	Total Core Damage Frequency (including uncategorized releases)	1.59E-05

5.1.1 Step One: Baseline Risk Determination

In this step, the baseline risk is determined. The example plant, VEGP, is a PWR with a large dry containment with the risk attributes provided in Table 5-1. The VEGP release categories, illustrated on Table 5-1, do not directly correspond to the EPRI accident classes. In addition, the VEGP release categories have an unclassified release category that requires classification in order to preserve total risk.

Table 5-2 provides the relationship between the EPRI accident class and the VEGP release categories. In addition, a scaling factor of 1.116 is used to apportion the unclassified release category evenly to the VEGP release categories. The scaling factor is determined by dividing the total core damage frequency (including the uncategorized frequency) by the total categorized release category frequency.

EPRI accident classes 4, 5, and 6 are not affected by the optimization of the ILRT testing interval and therefore are not included in the evaluation.

The accident bin frequencies for classes 3a and 3b are determined by multiplying the intact accident bin by the class 3a leakage probability and the class 3b leakage probability. The class 3a leakage probability is based on data from the ILRT testing data (Section 3.5), which is two "small" failures in 217 tests (2/217=0.0092).

The class 3b failure probability is based on the Jeffreys Non-Informative Prior and is equal to 0.0023 (Section 3.5).

Class 3a Frequency = CDF * Class 3a Leakage Probability

Class 3b Frequency = CDF * Class 3b Leakage Probability

Table 5-2
EPRI Accident Classes and Corresponding VEGP Release Category

EPRI Accident Class	VEPG Release Category	VEPG Release Category Definition	VEGP Release Category Frequency	Adjusted Frequency (factor 1.116)	EPRI Accident Class Frequency
1	A	No containment failure within 48-hour mission time, but failure could eventually occur without further mitigating action; noble gases and less than 0.1% volatiles released	1.42E-05	1.58E-05	1.58E-05
	S	Success (leakage only, success maintenance of containment integrity; containment not bypassed; isolation successful prior to core damage)	4.32E-09	4.82E-09	
2	G	Containment failure prior to vessel failure with noble gases and up to 10% volatiles released (containment not isolated)	5.98E-10	6.67E-10	6.67E-10
7	K	Late containment failure with noble gases and less than 0.1% volatiles released (containment failure greater than 6 hours after vessel failure; containment not bypassed; isolation successful prior to core damage)	2.23E-08	2.49E-08	2.49E-08
8	D	Containment bypassed with noble gases and up to 10% of the volatiles released	4.26E-09	4.75E-09	6.50E-08
	T	Containment bypassed with noble gases and more than 10% of the volatiles released	5.40E-08	6.03E-08	
		Total Frequency	1.42E-05	1.59E-05	1.59E-05

Supplemental guidance to the NEI Interim Guidance [20] provides additional information concerning the conservatisms in the quantitative calculation of delta LERF. The supplemental guidance describes methods, using plant-specific calculations to address the conservatisms. The supplemental guidance 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 the 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."

In the case of the VEGP, this translated to the removal of Class 1 individual sequences where containment sprays were available and VEGP Class 2 and 8. The individual sequences where containment spray is available can be removed due to the fact that a large release is very unlikely in these scenarios. The portion of class 1 where containment sprays were available is 2.35% of the total class 1 sequences. Classes 2 and 8 already result in LERF and therefore are unaffected by the change in ILRT testing interval.

Class 3a Frequency = (CDF – (0.0235 * Class 1) – Class 2 – Class 8) * 0.0092

Class 3a Frequency = (1.59E-05 – (0.0235 * 1.58E-05) – 6.67E-10 – 6.50E-08) * 0.0092

Class 3a Frequency = 1.42E-07 per year

Class 3b Frequency = (CDF – (0.0235 * Class 1) – Class 2 – Class 8) * 0.0023

Class 3b Frequency = (1.59E-05 – (0.0235 * 1.58E-05) – 6.67E-10 – 6.50E-08) * 0.0023

Class 3b Frequency = 3.56E-08 per year

Subtracting class 3a and class 3b frequencies from class 1 will preserve the total CDF. Therefore, the revised class 1 CDF is given as:

Class 1 Frequency (revised) = Class 1 Frequency – (Class 3a Frequency + Class 3b Frequency)

Class 1 Frequency (revised) = 1.58E-05 – (1.42E-07 + 3.56E-08)

Class 1 Frequency (revised) = 1.56E-05 per year

Table 5-3 presents a summary of the final VEGP frequencies for the EPRI Accident Classes. EPRI Accident Classes 4, 5, and 6 are omitted from the summary since these accident classes do not impact the calculation of the risk metrics of interest (see Section 4.3).

Table 5-3
VEGP EPRI Accident Class Frequencies

EPRI Accident Class	VEGP Frequency
1	1.56E-05
2	6.67E-10
3a	1.42E-07
3b	3.56E-08
7	2.49E-08
8	6.50E-08

5.1.2 Step Two: Develop the Baseline Population Dose

In this step, the baseline population dose (person-rem, from the plant-specific PRA or calculated based on leakage) is developed for the applicable accident classes.

In this example, the population dose is calculated by using the data provided in NUREG/CR-4551 [21] for the Surry Plant and by adjusting the results for VEGP. Specifically, each VEGP release category is associated with an applicable collapsed accident progression bin of NUREG/CR-4551. Table 5-4 provides a description of the collapsed accident progression bins (APBs) from NUREG/CR-4551.

The population dose risk at 50 miles is calculated for Surry for each of the accident progression bins. Table 5-5 provides the calculation of the Surry population dose risk at 50 miles for each of the accident progression bins.

Table 5-6 relates the VEGP release category with NUREG/CR-4551 accident progression bin and EPRI accident class.

Table 5-7 provides the resultant VEGP population dose for the EPRI accident classes of interest.

Table 5-4
Summary Accident Progression Bin (APB) Descriptions (NUREG/CR-4551, Surry)

Summary APB	Description
1	CD, VB, Early CF, Alpha Mode Core damage occurs, followed by a very energetic molten fuel-coolant interaction in the vessel; the vessel fails and generates a missile that fails the containment as well. Includes accidents that have an Alpha mode failure of the vessel and the containment, except those that follow Event V or an SGTR. It includes Alpha mode failures that follow isolation failures because the Alpha mode containment failure is of rupture size.
2	CD, VB, Early CF, RCS Pressure >200 psia Core damage occurs followed by vessel breach. Implies early CF with the RCS above 200 psia when the vessel fails. Early CF means at or before VB, so it includes isolation failures and seismic containment failures at the start of the accident as well as containment failure at VB. It does not include bins in which containment failure at VB follows Event V or an SGTR, or Alpha mode failures.
3	CD, VB, Early CF, RCS Pressure < 200 psia Core damage occurs, followed by vessel breach. Implies early CF with the RCS below psia when the containment fails. It does not include bins in which the containment failure is at VB or an SGTR, or Alpha mode failures.
4	CD, VB, Late CF Core damage occurs, followed by vessel breach. Includes accidents in which the containment was not failed or bypassed before the onset of core-concrete interaction (CCI) and in which the vessel failed. The failure mechanisms are hydrogen combustion during CCI, Basemat Melt-Through (BMT) in several days, or eventual overpressure due to the failure to provide containment heat removal in the days following the accident.
5	CD, Bypass Core damage occurs, followed by vessel breach. Includes Event V and SGTRs no matter what happens to the containment after the start of the accident. It also includes SGTRs that do not result in VB.
6	CD, VB, No CF Core damage occurs, followed by vessel breach. Includes accidents not evaluated in one of the previous bins. The vessel's lower head is penetrated by the core, but the containment does not fail and is not bypassed.
7	CD, No VB Core damage occurs but is arrested in time to prevent vessel breach. Includes accident progressions that avoid vessel failures, except those that bypass the containment. Most of the bins replaced in this reduced bin have no containment failures as well as no VB. It also includes bins in which the containment is not isolated at the start of the accident and in which the core is brought to a safe, stable state before the vessel fails.

Table 5-5
Calculation of Surry Population Dose Risk at 50 Miles

Collapsed APB	Fractional APB Contributions to Risk (MFCR) [(1)]	NUREG/CR-4551 Population Dose Risk at 50 Miles (person-rem/yr – mean) [(2)]	NUREG/CR-4551 Collapsed APB Frequency (per year) [(3)]	NUREG/CR-4551 Population Dose at 50 miles (person-rem) [(4)]
1	0.029	0.158	1.23E-07	1.28E+06
2	0.019	0.106	1.64E-07	6.46E+05
3	0.002	0.013	2.01E-08	6.46E+05 [(5)]
4	0.216	1.199	2.42E-06	4.95E+05
5	0.732	4.060	5.00E-06	8.12E+05
6	0.001	0.006	1.42E-05	4.23E+02
7	0.002	0.011	1.91E-05	5.76E+02
Totals	1.000	5.55	4.1E-05	

Notes:

(1) Mean Fractional Contribution to Risk calculated from the average of two samples delineated in Table 5.1-3 of NUREG/CR-4551.

(2) The total population dose risk at 50 miles from internal events in person-rem is provided as the average of two samples 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 2.5-3. 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.

(5) Assumed population dose at 50 miles from collapsed bin 3 is equal to collapsed bin 2. Collapsed bin 23 was back-calculated using that value. This does not influence the results of this evaluation since bin 3 does not appear as part of the results for VEGP.

Table 5-6 provides the VEGP-specific release categories and their association with NUREG/CR-4551 collapsed accident progression bins and EPRI accident classes.

Table 5-6
VEGP Release Category Application to NUREG/CR-4551 Accident Progression Bin and EPRI Accident Class

VEPG Release Category	VEPG Definition	NUREG/ CR-4551 APB	EPRI Accident Class
A	No containment failure within 48-hour mission time, but failure could eventually occur without further mitigating action; noble gases and less than 0.1% volatiles released	6	1
D	Containment bypassed with noble gases and up to 10% of the volatiles released	5	8
G	Containment failure prior to vessel failure with noble gases and up to 10% volatiles released (containment not isolated)	2	2
K	Late containment failure with noble gases and less than 0.1% volatiles released (containment failure greater than 6 hours after vessel failure; containment not bypassed; isolation successful prior to core damage)	4	7
S	Success (leakage only, success maintenance of containment integrity; containment not bypassed; isolation successful prior to core damage)	7	1
T	Containment bypassed with noble gases and more than 10% of the volatiles released	5	8

To determine the applicable population dose for VEGP, the population dose for the Surry collapsed accident progression bins (APBs) is used. The Surry population dose is adjusted for the VEGP plant-specific population using a “population dose factor.” The population dose factor is used to adjust the Surry population dose to account for changes in the population within the 50-mile radius of VEGP. The population dose factor is calculated by dividing the VEGP population by the Surry population information given in NUREG/CR-6441.

Total VEGP Population (50 miles) = 6.45E+05

Surry Population (NUREG/CR-6441) = 1.23E+06

Population Dose Factor = 6.45E+05 / 1.23E+06 = 0.524

The relationship above implies that the resultant doses are a direct function of population within 50 miles of each site. This does not take into account plant-specific differences in allowable leak rate, La, reactor power level, or other factors but does provide a reasonable first-order approximation of the population dose associated with NUREG/CR-4551 accident progression bins. A more in-depth adjustment of the population dose would include factors for rated reactor power level (the facility power level divided by the reference plant power level, MWt) and the allowable containment leak rate (La for the facility divided by La for the reference plant in %w/o/day). The leakage factor should be applied only to the APB that is influenced by normal leakage. The information in the BWR example (Section 5.2.2) and Appendix H, Section 4.2 provides a more in-depth treatment of accounting for the differences between the NUREG-4451 plant and the facility of interest.

Table 5-7 presents the VEGP population dose for the EPRI accident classes, excluding classes 3a and 3b. The data on the table were developed by re-sorting the information in Table 5-6 by EPRI accident class, adding the adjusted VEGP release category frequencies (Table 5-2), and accounting for the difference in population within a 50-mile radius of VEGP.

Table 5-7
VEGP Population Dose for EPRI Accident Classes

EPRI Accident Class	NUREG/ CR-4551 APB	VEGP Release Category Designator	VEGP Release Category Frequency	NUREG/CR-4551 Population Dose (50 miles) (person-rem)	Population Dose Factor	VEGP Population Dose
1	6	A	1.58E-05	4.23E+02	0.524	2.22E+02
	7	S	4.82E-09	5.76E+02	0.524	3.02E+02
2	2	G	6.67E-10	6.46E+05	0.524	3.39E+05
7	4	K	2.49E-08	4.95E+05	0.524	2.59E+05
8	5	D	4.75E-09	8.12E+05	0.524	4.25E+05
		T	6.03E-08			

To determine the dose rates for EPRI accident classes 3a and 3b, the population dose for EPRI accident class 1 (assumed to be 1 La) is multiplied by the factors of 10 La and 100 La, respectively. In the case of VEGP, a frequency-weighted dose is used to represent EPRI accident class 1 dose since the class is composed of multiple VEGP release categories. VEGP release categories A and S comprise EPRI accident class 1. The VEGP population dose for EPRI accident class 1 is calculated as:

The frequency-weighted fraction contribution of release category A:

> Release Category A Frequency / (Release Category A + Release Category S) * Release Category A Population Dose
>
> 1.58E-05/(1.58E-05 + 4.82E-09) * 2.22E+02
>
> = 2.22E+02

Plus the frequency-weighted fraction contribution of release category S:

Release Category S Frequency/(Release Category A + Release Category S) * Release Category S Population Dose

4.82E-09/(1.58E-05 + 4.82E-09) * 3.02E+02

= 8.18E-02

The frequency-weighted average population dose for the VEGP equivalent EPRI accident class 1 is determined by summing the contributions from VEGP release categories A and S. Due to the very low frequency contribution of VEGP release category S, the frequency weighted population dose for the equivalent EPRI accident class1 is equal to the population dose for VEGP release category A of 2.22E+02 person-rem.

Table 5-8
Population Dose for VEGP EPRI Accident Classes 3a and 3b

EPRI Accident Class	VEGP Frequency (per year)	EPRI Accident Class Leakage Rate	VEGP Population Dose
3a	1.42E-07	10 La	2.22E+03
3b	3.56E-08	100 La	2.22E+04

5.1.3 Step Three: Evaluate the Risk Impact (Bin Frequency and Population Dose)

In this step, the risk impact associated with the change in ILRT testing intervals is evaluated in terms of changes to the accident class frequencies and populations doses. This is accomplished in a three-step process.

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

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

In the third step, the percentile increase in dose rate for each extended interval is determined as follows: Percent increase = [(total dose rate of new interval minus total baseline dose rate) divided by (total baseline dose rate)] x 100.

Table 5-9
VEGP Accident Class Frequency and Population Doses as a Function of ILRT Frequency

EPRI Accident Class	Population Dose (person-rem)	ILRT Frequency					
		3 per 10 years		1 per 10 years		1 per 15 years	
		Frequency (per year)	Person-Rem/yr	Frequency (per year)	Person-Rem/yr	Frequency (per year)	Person-Rem/yr
1	2.22E+02	1.56E-05	3.47E-03	1.52E-05	3.38E-03	1.49E-05	3.31E-03
2	3.39E+05	6.67E-10	2.26E-04	6.67E-10	2.26E-04	6.67E-10	2.26E-04
3a	2.22E+03	1.42E-07	3.16E-04	4.74E-07	1.05E-03	7.11E-07	1.58E-03
3b	2.22E+04	3.55E-08	7.89E-04	1.18E-07	2.63E-03	1.78E-07	3.95E-03
7	2.59E+05	2.49E-08	6.45E-03	2.49E-08	6.45E-03	2.49E-08	6.45E-03
8	4.25E+05	6.50E-08	2.76E-02	6.50E-08	2.76E-02	6.50E-08	2.76E-02

5.1.4 Step Four: Evaluate Change in LERF and CCFP

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

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

ΔLERF = (frequency of class 3b new interval x) - (frequency of class 3b baseline)

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

CCFP = [1 - (frequency that results in no containment failure) / CDF] * 100%

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

CCFP Change (increase) = (CCFP at interval x) – (CCFP at baseline interval), expressed as percentage point change.

Table 5-10
VEGP Delta LERF and CCFP

Risk Metric	ILRT Testing Frequency		
	3 in 10 years	**1 in 10 years**	**1 in 15 years**
ΔLERF	N/A	8.29E-08	1.42E-07
ΔCCFP	N/A	0.52%	0.89%

5.1.5 Step Five: Evaluate Sensitivity of Results

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

In evaluating the impact of liner corrosion on the extension of ILRT testing intervals, the Calvert Cliffs methodology [17] is used. The methodology developed for Calvert Cliffs investigates how an age-related degradation mechanism can be factored into the risk impact associated with longer ILRT testing intervals.

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

5.1.5.1 Steel Liner Corrosion Sensitivity

This sensitivity study presents an estimate of the likelihood and risk implications of corrosion induced leakage of steel containment liners being undetected during the extended ILRT test intervals evaluated in this report. The methodology employed in this sensitivity case is taken from the Calvert Cliffs liner corrosion analysis. It is important to note that the corrosion analysis is a sensitivity case that represents the first 15-year extension. It is possible that for some slow corrosion mechanisms, such as embedment of debris in containment during initial containment construction, the probability of leakage can continue to increase over longer periods. However, these mechanisms are generally very slow and have a very limited potential for the development of large leakage pathways before detection.

The Calvert Cliffs analysis is performed for a concrete cylinder and dome with a concrete basemat, each with a steel liner. VEGP has a similar containment type.

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

- Differences between the containment basemat and the containment cylinder and dome
- The historical steel liner flaw likelihood due to concealed corrosion

- The impact of aging
- The corrosion leakage dependency on containment pressure
- The likelihood that visual inspections will be effective at detecting a flaw

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

- A half failure is assumed for the basemat concealed liner corrosion due to lack of identified failures.
- Two corrosion events are used to estimate the liner flaw probability. These events, one at North Anna Unit 2 and the other at Brunswick Unit 2, were initiated from the non-visible (backside) portion of the containment liner.
- The estimated historical flaw probability is limited to 5.5 years to reflect the years since September 1996 when 10CFR50.55a started requiring visual inspections. Additional success data were not used to limit the aging impact of the corrosion issue, even though inspections were being performed prior to this data (and have been performed since the timeframe of the Calvert Cliffs analysis) and there has been no evidence that additional corrosion issues were identified.
- The likelihood of the containment atmosphere reaching the outside atmosphere given that a liner flaw exists was estimated as 1.1% for the cylinder and dome and 0.11% (10% of the cylinder failure probability) for the basemat. These values were determined from an assessment of the probability versus containment pressure that corresponds to the ILRT pressure of 37 psig. For VEGP, the containment failure probabilities are less than these values at 37 psig. Conservative probabilities of 1% and 0.1% are used for the cylinder and dome and basemat, respectively.
- The likelihood of leakage escape (due to crack formation) in the basemat region is considered to be less than that in the containment cylinder and dome region.
- A 5% visual inspection detection failure likelihood, given that the flaw is visible, and a total detection failure likelihood of 10% are used. To date, all liner corrosion events have been detected through visual inspection.
- All non-detectable failures are assumed to result in large early releases. This approach is conservative and avoids detailed analysis of containment failure timing and operator recovery actions. That is, the probability of all non-detectable failures from the corrosion sensitivity analysis are added to the EPRI Class 3b (and subtracted from EPRI Class 1).

Table 5-11
VEGP Liner Corrosion Analysis

Step	Description	Containment Cylinder and Dome	Containment Basemat
1	Historical Steel Liner Flaw Likelihood	Events: 2	Events: 0 (assume half a failure)
	Failure Data:[1]	2 / (70 * 5.5) = 5.2E-3	0.5 / (70 * 5.5) = 1.3E-3
2	Age-Adjusted Steel Liner Flaw Likelihood [2]	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 [3a]	0.71% (1 to 3 years) 4.06% (1 to 10 years) 9.40% (1 to 15 years) [3b]	0.18% (1 to 3 years) 1.02% (1 to 10 years) 2.35% (1 to 15 years) [3c]
4	Likelihood of Breach in Containment Given Steel Liner Flaw [4]	1%	0.1%
5	Visual Inspection Detection Failure Likelihood	10% [5a]	100% [5b]
6	Likelihood of Non-Detected Containment Leakage (Steps 3*4*5)	0.00071% (at 3 years) 0.71% *1% * 10% 0.0041% (at 10 years) 4.1% * 1% * 10% 0.0094% (at 15 years) 9.4% * 1% * 10%	0.00018% (at 3 years) 0.18% * 0.1% * 100% 0.0010% (at 10 years) 1.0% * 0.1% * 100% 0.0024% (at 15 years) 2.4% * 0.1% * 100%

Notes:

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

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

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

(b) Note that the Calvert Cliffs analysis presents the delta between three and 15 years of 8.7% to utilize in the estimation of the delta-LERF value. For this analysis, however, the values are calculated based on three-, 10-, and 15-year intervals, consistent with the desired presentation of the results.

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

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

(5) (a) Five percent failure to identify visual flaws plus 5% likelihood that the flaw is not visible (not through cylinder but could be detected by ILRT). All events have been detected through visual inspection. Five percent visible failure detection is a conservative assumption.

(b) Cannot be visually inspected.

Table 5-12 provides a summary of the VEGP base case as well as the corrosion sensitivity case. The table is divided into three columns representing the frequency of the ILRT: Base Case (three per 10 years), one per 10 years, and one per 15 years.

Each of the three columns is sub-divided further into corrosion and non-corrosion cases. For both the corrosion and non-corrosion cases, the frequencies of the EPRI accident classes are provided. In the non-corrosion cases, an additional column titled “Delta person-rem per yr” is provided. The “Delta person-rem per yr” column provides the change in person-rem per year between the case corrosion and non-corrosion. Negative values in the “Delta person-rem per yr” column indicate a reduction in the person-rem per year for the selected accident class. This occurs only in EPRI accident class 1 and is a result of the reduction in the frequency of the accident class 1 and an increase in accident class 3b.

A row for the totals, both frequency and dose rate, is provided on the table. Additional summary rows are also provided.

- The change in dose rate (person-rem/year) and change as % of base total dose rate is provided below the “total” row.
- The Conditional Containment Failure Probability (CCFP) is provided in the next row.
- The percentage point change in CCFP from the base case (ΔCCFP) is provided in the next row
- Class 3b LERF is also provided and indicates the accident class 3b frequency as well as the change in the class 3b frequency in parentheses. This difference is calculated between the non-corrosion and corrosion cases.
- The next row, titled “Delta LERF from Base Case (three per 10 years),” provides the change in LERF as a function of ILRT frequency from the base case. The difference between the non-corrosion and corrosion cases is provided in parentheses .
- The last row of the table, titled “Delta LERF from One per 10 Years” provides the change in LERF as a result of changing the ILRT frequency from one in 10 years to one in 15 years. The difference between the non-corrosion and corrosion cases is provided in parentheses.

The sensitivity analysis of this section presents an estimate of the likelihood and risk implications of corrosion-induced leakage of steel containment liners not being detected during the extended ILRT test intervals evaluated in this report. The analysis considers ILRT extension time, inspections, and concealed degradation in uninspectable areas. As can be seen from Table 5-12, the change from the base case of three tests per 10 years to one test per 15 years with corrosion in LERF is very small, 1.44E-07 (the change in LERF for the same period without corrosion was 1.42E-07). The change in delta-LERF between the 15-year corrosion and non-corrosion cases is correspondingly very small, 1.8E-09.

Table 5-12
VEGP Summary of Base Case and Corrosion Sensitivity Cases

EPRI Class	Base Case (3 per 10 years)					1 per 10 years					1 per 15 years				
	Without Corrosion		With Corrosion			Without Corrosion		With Corrosion			Without Corrosion		With Corrosion		
	Frequency (per year)	Person-rem per year	Frequency (per year)	Person-rem per year	Delta person-rem per yr	Frequency (per year)	Person-rem per year	Frequency (per year)	Person-rem per year	Delta person-rem per yr	Frequency (per year)	Person-rem per year	Frequency (per year)	Person-rem per year	Delta person-rem per yr
1	1.56E-05	3.47E-03	1.56E-05	3.47E-03	-3.04E-08	1.52E-05	3.38E-03	1.52E-05	3.38E-03	-1.74E-07	1.49E-05	3.31E-03	1.49E-05	3.31E-03	-4.03E-07
2	6.67E-10	2.26E-04	6.67E-10	2.26E-04	0.00E+00	6.67E-10	2.26E-04	6.67E-10	2.26E-04	0.00E+00	6.67E-10	2.26E-04	6.67E-10	2.26E-04	0.00E+00
3a	1.42E-07	3.16E-04	1.42E-07	3.16E-04	0.00E+00	4.74E-07	1.05E-03	4.74E-07	1.05E-03	0.00E+00	7.11E-07	1.58E-03	7.11E-07	1.58E-03	0.00E+00
3b	3.55E-08	7.89E-04	3.57E-08	7.92E-04	3.04E-06	1.18E-07	2.63E-03	1.19E-07	2.65E-03	1.74E-05	1.78E-07	3.95E-03	1.80E-07	3.99E-03	4.03E-05
7	2.49E-08	6.45E-03	2.49E-08	6.45E-03	0.00E+00	2.49E-08	6.45E-03	2.49E-08	6.45E-03	0.00E+00	2.49E-08	6.45E-03	2.49E-08	6.45E-03	0.00E+00
8	6.50E-08	2.76E-02	6.50E-08	2.76E-02	0.00E+00	6.50E-08	2.76E-02	6.50E-08	2.76E-02	0.00E+00	6.50E-08	2.76E-02	6.50E-08	2.76E-02	0.00E+00
Total	1.59E-05	3.89E-02	1.59E-05	3.89E-02	3.01E-06	1.59E-05	4.14E-02	1.59E-05	4.14E-02	1.73E-05	1.59E-05	4.31E-02	1.59E-05	4.32E-02	3.99E-05
Δ Dose	N/A		N/A			2.49E-03 6.4%		2.50E-03 6.4%			4.26E-03 11.0%		4.30E-03 11.1%		
CCFP	0.79%		0.79%			1.32%		1.32%			1.69%		1.70%		
Δ CCFP	N/A		N/A			0.52%		0.53%			0.89%		0.91%		
Class 3b LERF	3.55E-08		3.57E-08 (1.4E-10)			1.18E-07		1.19E-07 (7.85E-10)			1.78E-07		1.80E-07 (1.8E-09)		
Delta LERF from Base Case (3 per 10 years)						8.2E-08		8.37E-08 (7.8E-10)			1.42E-07		1.44E-07 (1.8E-9)		
Delta LERF from 1 per 10 years						N/A					5.93E-08		6.03E-08 (1.0E-9)		

5.1.5.2 Potential Impacts from External Events

In the Vogtle Individual Plant Examination of External Events (IPEEE), the dominant risk contributor from external events is fire. Other external hazards, such as seismic and high winds, were found to be within acceptable limits. At the time of the IPEEE Vogtle internal events, CDF was 4.45E-05 per reactor year and the calculated fire CDF was 1.01E-05 per year. A fire LERF was not calculated.

The fire analysis is dominated by loss of offsite power sequences. The high-risk fire areas included those associated with the main control room, switchgear rooms, and other areas affecting electrical power supply and control (electrical raceways, cable spreading and electrical penetration rooms) in which a fire could lead to a station blackout, causing loss of reactor coolant pump seal cooling and core uncovery as a result of a seal loss of coolant.

Since the IPEEE, the Vogtle PRA has been updated several times. Loss of offsite power is no longer the dominant contributor, and the total CDF has dropped to 1.59E-05 per year. It is assumed that the external event CDF is approximately equal to the current PRA internal events CDF, given that the IPEEE fire analysis CDF was 1.01E-05 per year.

In this analysis, the total LERF (including aging and corrosion effects) is 2.46E-07 (classes 2, 3b, and 8 from Table 5-12). It is likely that the total LERF as a result of external events is much lower given that some LERF events that contribute directly to LERF—such as Interfacing System Loss of Coolant Accidents (ISLOCA) and Steam Generator Tube Ruptures (SGTR)—are not initiated or generally result from fire events. Conservatively assuming the LERF for external events is equal to that of internal events gives a total LERF of 4.91E-07 per year. This value is much lower than the Regulatory Guide 1.174 acceptance guideline of 1E-05 per year.

5.1.6 Summary of PWR Example Results

In summary, the change in risk associated with the extension of the ILRT testing interval for VEGP is small. Table 5-12 and the following paragraphs summarize the results of the evaluation.

A comparison of the base annual population dose (person-rem/yr) with previously approved submittals indicates that VEGP has an extremely small initial dose rate of 0.0389 person-rem/yr. The annual population dose for a one-in-10-years ILRT testing frequency is 0.0414 person-rem/yr and for a one-in-15-years ILRT testing frequency is 0.0431 person-rem/yr. Both of these ILRT intervals result in an extremely small annual population dose.

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 in the CDF below 10^{-6} year and LERF below 10^{-7} per year. “Small” changes are defined as LERF below 1E-05 per year, and the change in LERF between 1E-06 and 1E-07 per year. Since changes to the ILRT testing interval do not impact CDF, the relevant criteria is LERF. The increase in LERF resulting the example change in ILRT testing frequency from three in 10 years to one in 15 years is very conservatively estimated as 1.42E-07 per year, with the corresponding LERF of 1.78E-7 per year. This is a “small” change. Regulatory Guide 1.174 also states that when the calculated increase in LERF is between 1E-07 to 1E-06 per year, applications will be considered only if it can be reasonably shown that the total LERF is less than 1E-05 per reactor year.. The total LERF (including aging and corrosion effects) is 2.46E-07 (classes 2, 3b, and 8). This value is much lower than the total LERF acceptance guideline of 1E-05 per year in Regulatory Guide 1.174. In addition, considering external events also results in a LERF equal to 4.91E-07 per year, which remains significantly lower than the Regulatory Guide 1.174 acceptance guideline of 1E-05 per year.

The increase in the conditional containment failure frequency from the three-in-10-years ILRT testing frequency to a one-in-15-years testing frequency is 0.91% from an initial value of 0.79% to 1.70%, including the effects of aging and corrosion. Considering the discussion of Section 2.2, the change is judged to be very small.

On the above basis, it can be concluded that changing the ILRT testing frequency to one in 15 years represents a very small change to the VEGP risk profile.

5.2 BWR Example

This example provides the details of the methodology applied to Columbia Generating Station (CGS) operated by Energy Northwest. Large portions of this example are adapted from the Columbia submittal [19].

The CGS total core damage frequency is 7.33E-06 per year, and LERF is 6.9E-07 per year. The CGS PRA the CDF is binned into plant damage states (PDSs). Table 5-13 provides a summary of the CGS level 2 results.

Table 5-13
CGS Level 1 and LERF Results

PDS Class	PDS Description	PDS ID	CDF	LERF
I – Transient and Small LOCA with Loss of RPV Injection Capability	Short-term TXU with loss of containment air	IA1	2.58E-08	2.78E-09
	Short-term TXU with offsite power available	IA2	7.32E-07	7.88E-08
	Long-term TXU for LOSP with 1 diesel	IA3	1.12E-07	3.55E-09
	Loss of containment heat removal with failure of HPCS	IB0	6.92E-07	0
	Loss of all ECCS due to flooding	IC	1.88E-07	1.88E-07
	Long-term TUV with offsite power available	IG	1.38E-06	1.08E-09
	Long-term TUV for LOSP with 1 diesel available	IH	1.80E-07	1.42E-10
II – Transient with loss of containment heat removal	Long-term TW with stuck-open PORV	IIB	8.11E-09	0
	Long-term TW	IID	1.11E-06	0
III – LOCAs	Reactor Vessel Rupture	IIIC	3.00E-07	2.31E-10
	Large LOCA with failure of containment suppression	IIIE	0	0
IV – ATWS	ATWS with vessel intact at time of core uncovery	IVBA	1.24E-07	1.24E-07
	ATWS with vessel intact at time of core uncovery	IVBL	6.25E-08	6.23E-08
V – LOCA (BOC)	LOCA outside Containment	V	1.57E-07	1.57E-07
VI – Station Blackout	Short-term (<2hr) dc power and ADS available	VIA1	9.75E-07	6.74E-08
	Long-term (>6hr) dc and ADS not available, stuck-open SRV	VIA2	3.72E-08	0
	Long-term (>6hr) dc power not available; HPCS recoverable with recovery of ac power	VIB1	1.03E-06	0
	Long-term (>6hr) dc power not available; HPCS not recoverable	VIB2	2.12E-07	0
Totals			7.33E-06	6.9E-07

Table 5-14
CGS Level 2 Results for Containment End States

Level 2 End State	Frequency (per year)	Percent CDF
Containment Intact	2.20E-06	30%
Containment Failure – Large Early Release (not scrubbed)	6.9E-07	9.3%
Containment Failure – Large Late Release (not scrubbed)	3.80E-06	52%
Containment Failure – Late Release (scrubbed)	6.4E-07	8.7%
Total	7.33E-06	100%

5.2.1 Step One: Baseline Risk Determination

In this step, the baseline risk is determined. The example plant, CGS, is a BWR with a Mark II containment with the risk attributes provided in Tables 5-13 and 5-14.

The CGS frequency of EPRI accident class 1 is equal to the frequency of those accident sequences where the containment is intact. From Table 5-14 this is 2.20E-06 per year.

The CGS frequency of EPRI accident class 2 is estimated by multiplying the conditional probability of containment isolation failure by the portion of sequences that are challenged. The CGS PDSs that have containment already failed or bypassed are IC, II, IIIE, IV, and V. Therefore, EPRI accident class 2 does not include these accident sequences. Therefore, the EPRI accident class 2 is calculated as follows:

= (CDF – (PDS IC + II + IIIE + IV + V)) * Conditional isolation failure probability

= (7.33E-06 – (1.88E-07 + 1.12E-06 + 0 + 1.87E-07 + 1.57E-07)) * 7.80E-04

= 4.43E-09 per year

By definition, EPRI accident classes 4, 5, and 6 are not affected by the extension of the ILRT testing interval and are therefore not addressed in this example.

The frequency of EPRI accident class 7 is the accident sequences where containment is failed as a result of severe accident phenomena. The frequency of this EPRI accident class is not affected by the ILRT testing interval. However, for the purposes of population dose calculation, the CGS frequency associated with this accident class is divided into three sub-categories that are given in Table 5-14.

These are:

- EPRI accident class 7a (Large, early, and not scrubbed) = 5.29E-07

  Note: The release frequency associated with active containment isolation failure (4.43E-9Iyr, EPRI Category 2) and containment bypass scenarios (1.57E-7lyr, EPRI Category 8) have been subtracted from the Table 2-2 total of 6.9E-7Iyr of this CGS bin to prevent double counting.
- EPRI accident class 7b (Large, late, and not scrubbed) = 3.80E-06
- EPRI accident class 7c (Large, late, and scrubbed) = 6.40E-07

The total CGS EPRI accident class 7 is then 4.97E-06 per year.

EPRI accident class 8 consists of accident sequences in which the containment is bypassed. In the case of CGS, this is equivalent to PDS class V.

The accident bin frequencies for classes 3a and 3b are determined by multiplying the total CDF by the class 3a leakage probability and the class 3b leakage probability. The class 3a leakage probability is based on data from the ILRT testing data (Section 3.5), which is two “small” failures in 217 tests (2/217=0.0092). The class 3b failure probability is based on the Jeffreys Non-Informative Prior and is equal to 0.0023 (Section 3.5).

Class 3a Frequency = CDF * Class 3a Leakage Probability

Class 3b Frequency = CDF * Class 3b Leakage Probability

However, supplemental guidance to the NEI Interim Guidance [20] provides additional information concerning the conservatisms in the quantitative calculation of delta LERF. The supplemental guidance describes methods, using plant-specific calculations, to address the conservatisms. The supplemental guidance 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 the 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.”

In the example of CGS, the calculation of the EPRI accident classes 3a and 3b is performed by multiplying the frequency of accident sequences that are affected by the ILRT testing interval extension by the conditional probability of failure. The frequency of accident sequences affected is equal to the total CDF minus those accident sequences that always result in LERF and those that never result in LERF regardless of ILRT testing frequency.

In the case of CGS, containment bypasses, internal flooding (that fails all ECCS), and ATWS accident sequences always result in LERF. Long-term station blackout and loss of containment heat removal accident sequences never result in LERF. Table 5-15 presents a summary of the CGS plant damage state classes that always or never result in LERF.

Table 5-15
CGS Accident Sequences for Consideration in EPRI Classes 3a and 3b

Plant Damage State Class	PDS ID	Frequency (per year)	Class Frequency (per year)
Accident Sequences Result in LERF			
Containment Bypass Accidents	V	1.57E-07	1.57E-07
Internal Flooding Accidents	IC	1.88E-07	1.88E-07
ATWS Accidents	IVBA	1.24E-07	1.87E-07
	IVBL	6.25E-08	
Accident Sequences Never Result in LERF			
Long-Term Station Blackout Accidents	VIA2	3.72E-08	1.28E-06
	VIB1	1.03E-06	
	VIB2	2.12E-07	
Loss of Containment Heat Removal	IIB	8.11E-09	1.12E-06
	IID	1.11E-06	
Totals		2.93E-06	2.93E-06

The CGS EPRI Accident Classes 3a and 3b can be calculated as follows:

Class 3a Frequency = (CDF – Always or Never LERF CDF) * Class 3a Leakage Probability

Class 3a Frequency = (7.33E-06 per year – 2.93E-06 per year) * 0.0092

Class 3a Frequency = 4.05E-08 per year

Class 3b Frequency = (CDF – Always or Never LERF CDF) * Class 3b Leakage Probability

Class 3b Frequency = (7.33E-06 per year – 2.93E-06 per year) * 0.0023

Class 3b Frequency = 1.01E-08 per year

Table 5-16 provides a summary of the CGS frequencies for the various EPRI accident classes of interest.

Table 5-16
CGS EPRI Accident Classes

EPRI Accident Class	CGS Frequency (per year)
1	2.15E-06
2	4.43 E-09
3a	4.05E-08
3b	1.01E-08
7	4.97E-6
8	1.57E-07

5.2.2 Step Two: Develop the Baseline Population Dose

The CGS population dose is calculated using the data provided in NUREG/CR-4551 for Peach Bottom and adjusting the results for applicability to CGS. Each Peach Bottom accident sequence was assigned to an applicable Accident Progression Bin in NUREG/CR-4551. The definitions of the Accident Progression Bins are provided in Table 5-17.

The Peach Bottom population doses are adjusted to account for several CGS specific differences. Specifically, the Peach Bottom doses are adjusted for population and reactor power level.

Population Adjustment

The population within a 50-mile radius of Peach Bottom used in the NUREG/CR-4551 is 3.2E+06 persons. The population within a 50-mile radius of CGS is estimated at 3.6E+05 persons. A ratio of the population between the two plants is given as:

Population of Columbia (50 miles) / Population of Peach Bottom (50 miles) =

3.6E+05 / 3.2E+06 = 0.11

Power Level Adjustment

The Peach Bottom power level used in NUREG/CR-4551 consequence analysis is 3293 MWt. The CGS power level is 3486 MWt. The CGS power level is a factor of 1.06 greater than Peach Bottom’s (3486 MWt / 3293 MWt).

Table 5-17
Peach Bottom (NUREG/CR-4551) Accident Progression Bin Definitions

Collapsed APB	Accident Progression Bin 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	CB, 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 PRV 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 the 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 the 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.
7	CD, VB, No CF, Vent, N/A Core damage occurs, followed by vessel breach. The containment never structurally fails but is vented some time 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.

Containment Allowable Leakage

The resultant population dose is a function of the allowable leakage since it is measured on a percentage by weight basis. Both Peach Bottom and Columbia have allowable leakages of 0.5 percent per day, therefore, no correction for leakage is necessary ($F_{leakage}$=1).

The factors developed above are used to adjust the population dose for the surrogate plant (Peach Bottom) for CGS. For intact containment end states, the total population dose factor is as follows:

$$F_{Intact} = F_{Population} * F_{Power\ Level} * F_{Leakage}$$

$$F_{Intact} = 0.11 * 1.06 * 1$$

$$F_{Intact} = 0.12$$

For EPRI accident classes not dependent on containment leakage, the population dose factor is as follows:

$$F_{Others} = F_{Population} * F_{Power\ Level}$$

$$F_{Others} = 0.11 * 1.06$$

$$F_{Others} = 0.12$$

The Peach Bottom population dose by accident progression bins is presented in Table 5-18. It should be noted that Table 5-18 is calculated from NUREG/CR-4551 documentation since NUREG/CR-4551 does not provide the population dose based on accident progression bin.

The dose for EPRI accident class is determined by associating the EPRI accident class with an accident progression bin or bins. In the case of EPRI accident class 1, the APB that most closely approximates an intact containment is bin #8.

The dose EPRI accident class 2 is associated with accident progression bin #3. This assignment is based on assuming that the containment isolation failure of EPRI accident class 2 occurs in the drywell as an unscrubbed release. APB #3 results in the highest dose of all the Peach Bottom containment failure APBs, which is indicative of an unscrubbed release.

Table 5-18
Peach Bottom and CGS Population Doses [1]

Collapsed APB	Collapsed APB Frequency (per year) [2]	Fractional APB Contributions to Risk (MFCR) [3]	Population Dose Risk (50 miles) (person-rem/yr) [4]	Population Dose (50 miles) (person-rem) [5]	Population Dose Factor	CGS Population Dose (50 mile) (person-rem)
1	9.55E-08	0.021	0.166	1.74E+06	0.12	2.03E+05
2	4.77E-08	0.0066	0.0521	1.09E+06	0.12	1.27E+05
3	1.48E-06	0.556	4.39	2.97E+06	0.12	3.46E+05
4	7.94E-07	0.226	1.79	2.25E+06	0.12	2.62E+05
5	1.30E-08	0.0022	0.0174	1.34E+06	0.12	1.56E+05
6	2.04E-07	0.059	0.466	2.28E+06	0.12	2.66E+05
7	4.77E-07	0.118	0.932	1.95E+06	0.12	2.27E+05
8	7.99E-07	0.0005	3.95E-03	4.94E+03	0.12	5.76E+02
9	3.85E-07	0.01	0.079	2.05E+05	0.12	2.39E+02
10	4.34E-08	0	0	0	0.12	0
Totals	4.34E-06	1	7.9	-	-	-

Notes:

(1) This table is presented in the form of a calculation because NUREG/CR-4551 does not document dose results as a function of accident progression bin. As such, the dose results as a function of APB must be calculated from documented APB frequencies and APB dose results.

(2) The total CDF of 4.34E-06 per year and the CDF subtotals by APB are taken from Figure 2.5-6 of NUREG/CR-4551, Volume 4, Revision 1, Part I.

(3) The individual APB contributions to the total 50-mile radius dose rate are taken from Table 5.2-3 of NUREG/CR-4551, Volume 4, Revision 1, Part I.

(4) The APB 50-mile dose rate is calculated by multiplying the individual APB dose rate fractional contributions (column 4) by the total 50-mile radius dose rate of 7.9 person-rem per year (taken from Table 5.1-1 of NUREG/CR-4551, Volume 4, Revision 1, Part I).

(5) The individual doses are calculated by dividing the individual APB dose rate (column 5) by the APB frequencies (column 3).

In the case of EPRI accident classes 3a and 3b, no association is made with the NUREG/CR-4551 APBs. Rather, in accordance with the methodology, these accident classes are assigned 10 La and 100 La, or 10 and 100 times the dose associated with EPRI accident class 1.

EPRI accident classes 4, 5, and 6 are not affected by the ILRT testing interval and are not included in this analysis.

The dose associated with EPRI accident class 7 is based on a frequency-weighted average person-rem dose representative of the EPRI accident sub-classes of 7a, 7b, and 7c. EPRI accident classes 7a, 7b, and 7c are associated with APBs numbers 3, 4, and 5. The EPRI accident class 7 population dose is calculated in Table 5-19. The population dose factor of 0.12 is applied to the Peach Bottom population doses.

Table 5-19
CGS EPRI Accident Class Population Doses

EPRI Accident Class	Peach Bottom APB	CGS PDS Frequency (1)	Peach Bottom Population Doses (2)	CGS Population Dose (50 miles) Person-rem (3)	CGS Population Dose Rate (50 miles) person-rem/yr (4)
7a	3	5.29E-07	2.97E+06	3.46E+05	1.83E-01
7b	4	3.80E-06	2.25E+06	2.62E+05	9.96E-01
7c	5	6.40E-07	1.34E+06	1.56E+05	9.98E-02
Total		4.97E-06		2.57E+05 (5)	1.28

Notes:

(1) Taken from Section 5.2.1

(2) Taken from Table 5-18

(3) Calculated by multiplying column 4 by population dose factor of 0.12

(4) Obtained by multiplying the release frequency (column 3) by the CGS population dose (column 5)

(5) Frequency-weighted average population dose for EPRI accident class 7 obtained by dividing total population dose rate (1.28 person-rem/year) by the total release frequency (4.97E-06 per year)

The CGS population dose for EPRI accident class 8 is assigned the highest of the dose rates associated with the Peach Bottom accident progression bins, APB 3. Table 5-20 provides a summary of the CGS population doses for the EPRI accident classes.

Table 5-20
CGS Population Doses for EPRI Accident Classes

EPRI Accident Class	Class Description	CGS Person-rem Within 50 Miles (1)	Revised CGS Frequency (2)	Dose Rate (person-rem/yr) (3)
1	No Containment Failure	5.76E+02	2.15E-06	1.24E-03
2	Containment Isolation Failure	3.46E+05	4.43 E-09	1.53E-03
3a	Small Pre-Existing Leak (4)	5.76E+03	4.05E-08	2.33E-04
3b	Large Pre-Existing Leak (5)	5.76E+04	1.01E-08	5.83E+04
7	Containment Failure - Severe Accident	2.57E+05	4.97E-6	1.28E+00
8	Containment Bypass	3.46E+05	1.57E-07	5.43E-02
Totals		1.03E06	7.33E-06	1.34

Notes:

(1) Population dose taken from Table 5-18

(2) Revised CGS frequency taken from Table 5-16

(3) Dose rate calculated by multiplying column 3 by column 4

(4) Pre-existing small leak population dose equal to 10 times EPRI accident class 1 population dose

(5) Pre-existing large leak population dose equal to 100 times EPRI accident class 1 population dose

5.2.3 Step Three: Evaluate the Risk Impact (Bin Frequency & Population Dose)

In this step, the risk impact associated with the change in ILRT testing intervals is evaluated in terms of changes to the accident class frequencies and populations doses. This is accomplished in a three-step process.

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

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

In the third step, the percentile increase in dose rate for each extended interval is determined as follows: Percent increase = [(total dose rate of new interval minus total baseline dose rate) divided by (total baseline dose rate)] x 100.

Table 5-21
CGS EPRI Accident Class Frequency and Population Doses as Functions of ILRT Frequency

EPRI Accident Class	Population Dose (person-rem)	ILRT Frequency					
		3 per 10 years		1 per 10 years		1 per 15 years	
		Frequency	Person-Rem/yr	Frequency	Person-Rem/yr	Frequency	Person-Rem/yr
1	5.76E+02	2.15E-06	1.24E-03	2.03E-06	1.17E-03	1.95E-06	1.12E-03
2	3.46E+05	4.43E-09	1.53E-03	4.43E-09	1.53E-03	4.43E-09	1.53E-03
3a	5.76E+03	4.05E-08	2.33E-04	1.35E-07	7.77E-04	2.02E-07	1.17E-03
3b	5.76E+04	1.01E-08	5.83E-04	3.37E-08	1.94E-03	5.06E-08	2.92E-03
7	2.57E+05	4.97E-06	1.28E+00	4.97E-06	1.28E+00	4.97E-06	1.28E+00
8	3.46E+05	1.57E-07	5.43E-02	1.57E-07	5.43E-02	1.57E-07	5.43E-02
Totals	1.03E+06	7.33E-06	1.34E+00	7.33E-06	1.34E+00	7.33E-06	1.34E+00

5.2.4 Step Four: Evaluate Change in LERF and CCFP

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

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

$$\Delta LERF = (\text{frequency of class 3b new interval x}) - (\text{frequency of class 3b baseline})$$

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

CCFP = [1 - (frequency that results in no containment failure) / CDF] * 100%

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

CCFP Change (increase) = (CCFP at interval x) – (CCFP at baseline interval), expressed as percentage point change.

Table 5-22
CGS Delta LERF and CCFP

	ILRT Testing Frequency		
	3 in 10 years	1 in 10 years	1 in 15 years
ΔLERF	N/A	2.36E-08	4.05E-08
ΔCCFP	N/A	0.32%	0.55%

5.2.5 Step Five: Evaluate Sensitivity of Results

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

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

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

5.2.5.1 Steel Liner Corrosion Sensitivity

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

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

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

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

- A half failure is assumed for the basemat concealed liner corrosion due to lack of identified failures.
- Two corrosion events are used to estimate the liner flaw probability. These events, one at North Anna Unit 2 and the other at Brunswick Unit 2, were initiated from the non-visible (backside) portion of the containment liner.
- The estimated historical flaw probability is limited to 5.5 years to reflect the years since September 1996, when 10CFR50.55a started requiring visual inspections. Additional success data were not used to limit the aging impact of the corrosion issue, even though inspections were being performed prior to this data (and have been performed since the timeframe of the Calvert Cliffs analysis) and there has been no evidence that additional corrosion issues were identified.
- Consistent with the Calvert Cliffs analysis, the corrosion-induced steel liner/shell flaw likelihood is assumed to double every five years. This is based solely on judgment and is included in this analysis to address the increase in likelihood of corrosion as the steel shell ages.
- The likelihood of the containment atmosphere reaching the outside atmosphere given that a liner flaw exists was estimated as 1.1% for the cylinder and dome and 0.11% (10% of the cylinder failure probability) for the basemat. These values were determined from an assessment of the probability versus containment pressure that corresponds to the ILRT pressure of 37 psig. For CGS, the containment failure probabilities are conservatively assumed to be 10% for the shell wall and 1% for the basemat. Since the basemat of CGS is the suppression pool, it is judged that a failure of the containment in this area would not lead to LERF. Hence, the assumed 1% probability is particularly conservative.

- In the Calvert Cliffs analysis, it is noted that approximately 85% of the interior wall surface is accessible for visual inspections. At CGS the interior wall surface accessible for visual inspections is estimated at 90% (the majority of the uninspectable wall surface being the area between the drywell floor slab and the DW-WW omega seal). Given that the flaw is visible, a 5% visual inspection detection failure likelihood and a total detection failure likelihood of 10% are used. To date, all liner corrosion events have been detected through visual inspection.
- All non-detectable failures are assumed to result in early releases. This approach is conservative and avoids detailed analysis of containment failure timing and operator recovery actions.

Table 5-23
CGS Liner Corrosion Analysis

Step	Description	Containment Walls	Containment Basemat
1	Historical Steel Liner Flaw Likelihood Failure Data:(1)	Events: 2 2/(70 * 5.5) = 5.2E-3	Events: 0 (assume 0.5 failure) 0.5/(70 * 5.5) = 1.3E-3
2	Age-Adjusted Steel Liner Flaw Likelihood (2)	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 (3a)	0.71% (1 to 3 years) 4.06% (1 to 10 years) 9.40% (1 to 15 years) (3b)	0.18% (1 to 3 years) 1.02% (1 to 10 years) 2.35% (1 to 15 years) (3c)
4	Likelihood of Breach in Containment Given Steel Liner Flaw (4)	1%	0.1%
5	Visual Inspection Detection Failure Likelihood	10% (5a)	100% (5b)
6	Likelihood of Non-Detected Containment Leakage (Steps 3*4*5)	0.00071% (at 3 years) 0.71% *1% * 10% 0.0041% (at 10 years) 4.1% * 1% * 10% 0.0094% (at 15 years) 9.4% * 1% * 10%	0.00018% (at 3 years) 0.18% * 0.1% * 100% 0.0010% (at 10 years) 1.0% * 0.1% * 100% 0.0024% (at 15 years) 2.4% * 0.1% * 100%

Notes:

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

2. During 15-year interval, assume failure rate doubles every five years (14.9% increase per year). The average for 5th to 10th year set to the historical failure rate (consistent with Calvert Cliffs analysis).

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

   (b) Note that the Calvert Cliffs analysis presents the delta between three and 15 years of 8.7% to utilize in the estimation of the delta-LERF value. For this analysis, however, the values are calculated based on three-, 10-, and 15-year intervals, consistent with the desired presentation of the results.

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

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

5. (a) 5% failure to identify visual flaws plus 5% likelihood that the flaw is not visible (not through-cylinder but could be detected by ILRT). All events have been detected through visual inspection. A 5% visible failure detection is a conservative assumption.

   (b) Cannot be visually inspected.

The cumulative likelihood of non-detected containment leak due to corrosion is the sum in step 6 for the containment walls and the containment basemat:

- At 3 years: 7.12E-05 + 1.78E-05 = 8.90E-05
- At 10 years: 4.14E-04 + 1.03E-04 = 5.17E-04
- At 15 years: 9.66E-04 + 2.41E-04 = 1.21E-03

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

- Per Table 5-21, the EPRI Class 3b frequency is 1.01E-08/yr.
- As discussed in Section 5.2.1, the CGS CDF associated with accidents that are not independently LERF or could never result in LERF is 7.33E-06/yr-2.93E-06/yr=4.4E-06/yr.
- The increase in the base case Class 3b frequency due to the corrosion-induced concealed flaw issue is calculated as 4.40E-06*8.90E-05=3.92E-10/yr, where 8.90E-5 was previously shown to be the cumulative likelihood of non-detected containment leakage due to corrosion at three years.
- The three-in-10-years Class 3b frequency, including the corrosion-induced concealed flaw issue, is calculated as 1.01E-08/yr+3.92E-10/yr=1.05E-08/yr.

Table 5-24 provides a summary of the base case as well as the corrosion sensitivity case. The table is divided into three columns representing the frequency of the ILRT: Base Case (three per 10 years), one per 10 years, and one per 15 years.

Each of the three columns is sub-divided further into corrosion and non-corrosion cases. For both the corrosion and non-corrosion cases, the frequencies of the EPRI accident classes are provided. In the non-corrosion cases, an additional column titled "Delta person-rem per yr" is provided. The "Delta person-rem per yr" column provides the change in person-rem per year between the corrosion and non-corrosion cases. Negative values in the "Delta person-rem per yr" column indicate a reduction in the person-rem per year for the selected accident class. This occurs only in EPRI accident class 1 and is a result of the reduction in the frequency of the accident class 1 and an increase in accident class 3b.

Rows for the totals, both frequency and dose rate, are provided in the table. Additional summary rows are also provided.

- The change in dose rate, expressed as person-rem/yr and percentage of the total base dose is provided in the row below the "total" row.
- The Conditional Containment Failure Probability (CCFP) is provided in the next row, followed by the change in CCPF in percentage points.
- Class 3b LERF is also provided and indicates the accident class 3b frequency as well as the change in the class 3b frequency in parentheses. This difference is calculated between the non-corrosion and corrosion cases.
- The next row titled "Delta LERF From Base Case (3 per 10 years)" provides the change in LERF as a function of ILRT frequency from the base case. The difference between the non-corrosion and corrosion cases is provided in parentheses.
- The last row of the table titled "Delta LERF From 1 per 10 Years" provides the change in LERF as a result of changing the ILRT frequency from one in 10 years to one in 15 years. The difference between the non-corrosion and corrosion cases is provided in parentheses.

The sensitivity analysis of this section presents an estimate of the likelihood and risk implications of corrosion-induced leakage of steel containment liners not being detected during the extended ILRT test intervals evaluated in this report. The analysis considers ILRT extension time, inspections, and concealed degradation in uninspectable areas. As can be seen from Table 5-24, the change from the base case of three tests per 10 years to one test per 15 years in LERF with corrosion is very small—4.6E-08. Similarly, the change in delta-LERF between the corrosion and non-corrosion cases for one per 15 years is correspondingly very small at 5.3E-09.

Table 5-24
Summary of CGS Base Case and Corrosion Sensitivity Cases

EPRI Class	Base Case (3 per 10 years)					1 per 10 years					1 per 15 years				
	Without Corrosion		With Corrosion			Without Corrosion		With Corrosion			Without Corrosion		With Corrosion		
	Frequency (per year)	Person-rem per year	Frequency (per year)	Person-rem per year	Delta person-rem per yr	Frequency (per year)	Person-rem per year	Frequency (per year)	Person-rem per year	Delta person-rem per yr	Frequency (per year)	Person-rem per year	Frequency (per year)	Person-rem per year	Delta person-rem per yr
1	2.15E-06	1.24E-03	2.15E-06	1.24E-03	-2.26E-07	2.03E-06	1.17E-03	2.03E-06	1.17E-03	-1.31E-06	1.95E-06	1.12E-03	1.94E-06	1.12E-03	-3.07E-06
2	4.43E-09	1.53E-03	4.43E-09	1.53E-03	0.00E+00	4.43E-09	1.53E-03	4.43E-09	1.53E-03	0.00E+00	4.43E-09	1.53E-03	4.43E-09	1.53E-03	0.00E+00
3a	4.05E-08	2.33E-04	4.05E-08	2.33E-04	0.00E+00	1.35E-07	7.77E-04	1.35E-07	7.77E-04	0.00E+00	2.02E-07	1.17E-03	2.02E-07	1.17E-03	0.00E+00
3b	1.01E-08	5.83E-04	1.05E-08	6.06E-04	2.26E-05	3.37E-08	1.94E-03	3.60E-08	2.07E-03	1.31E-04	5.06E-08	2.92E-03	5.59E-08	3.22E-03	3.07E-04
7	4.97E-06	1.28E+00	4.97E-06	1.28E+00	0.00E+00	4.97E-06	1.28E+00	4.97E-06	1.28E+00	0.00E+00	4.97E-06	1.28E+00	4.97E-06	1.28E+00	0.00E+00
8	1.57E-07	5.43E-02	1.57E-07	5.43E-02	0.00E+00	1.57E-07	5.43E-02	1.57E-07	5.43E-02	0.00E+00	1.57E-07	5.43E-02	1.57E-07	5.43E-02	0.00E+00
Total	7.33E-06	1.34E+00	7.33E-06	1.34E+00	2.23E-05	7.33E-06	1.34E+00	7.33E-06	1.34E+00	1.30E-04	7.33E-06	1.34E+00	7.33E-06	1.34E+00	3.04E-04
Δ Dose %	N/A						1.84E-03 0.14%		1/94E-03 0.15%			3.15E-03 0.24%		3.43E-03 0.26%	
CCFP	70.1%		70.1%			70.4%		70.5%			70.7%		70.8%		
ΔCCFP	N/A					0.32%		0.35%			0.55%		0.62%		
Class 3b LERF	1.01E-08		1.05E-08 (4E-10)			3.37E-08		3.60E-08 (2..3E-09)			5.06E-08		5.59E-08 (5.3E-09)		
Delta LERF (from base case of 3 per 10 years)						2.36E-08		2.59E-08 (2.3E-09)			4.05E-08		4.58E-08 (5.3E-09)		
Delta LERF from 1 per 10 years						N/A					1.69E-08		1.998E-08 (3.1E-09)		

5.2.5.2 Potential Impacts from External Events

External events were evaluated in the CGS Individual Plant Examination of External Events (IPEEE). The IPEEE program was a one-time review of external hazard risk and was limited in its purpose to the identification of potential plant vulnerabilities and an understanding of severe accident risk.

The primary areas of external event analysis for the CGS IPEEE were seismic hazards, internal fires, and volcanic activity. Adequate assurance regarding safe shutdown for volcanic events (that is, design basis ash fall) was addressed via plant procedures and equipment modifications, and no further examination (that is, quantitative assessment) was performed for the IPEEE.

Seismic events were addressed through a Seismic Probabilistic Safety Assessment (SPSA) as part of the IPEEE. The seismic external event study provides adequate (but conservative) information to assess the impact of seismic hazards on the conclusions of the CGS ILRT interval extension risk assessment.

Internal fire events were addressed through a Fire Probabilistic Safety Assessment (FPSA). Its conclusions are considered a reasonable reflection of the current state of the technology and adequate for assessing the impact of fires on the conclusions of the ILRT interval extension risk assessment.

The CGS fire PRA was updated in 2003, and the CDF contribution due to fire events is 1.08E-05 per year. As part of the impact assessment on possible large early releases, the CGS FPSA coupled with available generic insights offers the following conclusions with regard to the impact of fire events on containment performance:

- The FPSA investigated fire-induced containment isolation failures and determined that scenarios with containment isolation were not likely containment failure modes.
- The FPSA does not quantify the LERF risk measure; however, a review of NUREG-1742, Perspectives Gained from the IPEEE Program, indicates that the fire CDF for BWRs is primarily determined by plant transient type of events.

Given the above, it is judged reasonable to assume that the ratio of LERF to CDF for fire events is comparable to the ratio determined for internal events. For CGS internal events, the ratio of LERF (6.90E-07 per year) to CDF (7.33E-06 per year) is approximately 9.4%. As such, it is reasonable to assume here that fire-induced LERF is approximately 10% of fire-induced CDF (1.08E-05 per year), or 1.1E-06 per year.

The CGS seismic PSA was performed as part of the IPEEE. The SPSA CDF is 2.1E-05 per year. The CGS IPEEE SPSA was developed as a screening tool for one-time use in resolving the Generic Letter 88-20 issues. As such, the CGS SPSA is not on the same level of realism as the internal events CDF. Similar to the CGS FPSA, the SPSA does not provide a detailed breakdown of the seismic risk profile by accident class. The CGS SPSA does not distinguish between LERF and non-LERF accident sequence end states. The following were applied to determine the LERF and non-LERF end states for the SPSA:

- An evaluation of the accident sequences to assess whether the timing of a projected release would be great than 4 hours following a declaration of a general emergency (GE). This evaluation determined that approximately 9% of the seismic CDF consists of core damage in the early timeframe. Conservatively assuming that all such seismic CDF accidents result in a large magnitude release, the CGS seismic LERF can be approximated as 1.9E-06 per year.
- An assessment of the ability to evacuate people was performed. This assessment assumed that for seismic accelerations of less than 0.3g, evacuation is similar to the internal events study. For seismic accelerations greater than 0.5g, evacuation was conservatively not credited. For seismic accelerations between 0.3g and 0.5g, it was assumed that these scenarios are non-LERF.

Other external events evaluated for CGS included volcanic activity, high winds/tornados, external flooding, transportation and nearby facility accidents and other hazards. The CGS IPEEE analysis of these hazards was accomplished by reviewing plant environs against established regulatory requirements. Based upon this review, it was concluded that CGS meets applicable regulatory requirements and therefore has an acceptable low risk with respect to these hazards. As such, these hazards were determined in the CGS IPEEE to be negligible contributors to overall risk. Accordingly, these hazards are not included explicitly in this analysis and are reasonably assumed not to impact the results or conclusion of the ILRT interval extension risk assessment.

Per the guidance contained in this report, the figure-of-merit for the risk impact assessment of extended ILRT intervals is given as:

delta LERF = The change in frequency of EPRI Accident Class 3b

Using the percentage of total CDF contributing to LERF for the fire and seismic external events as an approximation for the early CDF applicable to EPRI Accident Class 3b yields the following:

$$\text{Class 3b Frequency} = [(CDF_{Fire} * 0.10) + (CDF_{Seismic} * 0.09)] * \text{Class 3b Leakage Probability}$$

$$\text{Class 3b Frequency} = [(1.08E\text{-}05 * 0.10) + (2.1E\text{-}05 * 0.09)] * 2.3E\text{-}03$$

$$\text{Class 3b Frequency} = 6.8E\text{-}09 \text{ per year}$$

Given the extremely conservative nature of the external events studies and the fact that many of the external event scenarios are long-term station blackout and long-term containment heat removal, use of the percentage is appropriate. Table 5-25 was developed using the relationships developed previously in the report for the LERF as a function of ILRT interval.

The case concerning external events contribution to the risk impact assessment of ILRT extensions is more compelling when conservative assumptions do not alter the conclusion. Therefore, in this case it would be prudent to use more conservative assumptions (such as all external event CDFs are applicable to class 3b) since the overall risk impact assessment conclusions would not change in this light.

Table 5-25
Upper Bound External Event Impact on ILRT LERF Calculation

Hazard	EPRI Accident Class 3b Frequency			LERF Increase (from 1 per 10 years)
	3 per 10 years	1 per 10 years	1 per 15 years	
External Events	6.8E-09	2.27E-08	3.41E-08	1.14E-08
Internal Events	1.01E-08	3.37E-08	5.06E-08	1.69E-08
Combined	1.69E-08	5.64E-08	8.47E-8	2.83E-08

5.2.6 Summary of BWR Example Results

NRC Regulatory Guide 1.174 provides NRC recommendations for using risk information in support of applications requesting changes to the license basis of the plant. The Regulatory Guide 1.174 acceptance guidelines are used here to assess the ILRT interval extension.

The calculated 2.83E-08 increase in LERF is due to the combined internal and external events from extending the ILRT testing frequency from one per 10 years to one per 15 years. Per Regulatory Guide 1.174, this is a “very small change” in risk. Considering the overall change in ILRT frequency from three in 10 years to one in 15 years results in a change in LERF of 6.8E-08 per year, which also falls into the “very small region” risk increase as defined by Regulatory Guide 1.174.

Per Regulatory Guide 1.174, when the calculated change in LERF is between 1E-07 and 1E-06 per year (that is, “small change” in risk), the assessment must also reasonably show that the total LERF remains below 1E-05 per year. While not required in this assessment, the total LERF is calculated for completeness. Table 5-26 was developed from previous analysis in the report.

Table 5-26
CGS Total LERF

Hazard	LERF Frequency
Fire	1.1E-06
Seismic	1.9E-06
Internal Events	6.9E-07
Total	3.7E-06

6
RESULTS SUMMARY AND CONCLUSIONS

This report section provides a summary of the results from the two example plants and draws conclusions from these examples as well as the approximately 59 submittals made to the NRC. See Appendix G for a summary of submittals.

6.1 Results Summary

Table 6-1 provides a summary of the important risk metrics for the ILRT interval extension for VEGP. The risk metric changes are presented for the base case and the sensitivity case performed. Tables 6-1 and 6-2 summarize the ILRT interval extension risk metrics for the two example plants. The only EPRI Accident Classes that are presented in summary Tables 6-1 and 6-2 are 3a and 3b. This is due to the fact that these are the accident classes that significantly impact the changes in the risk metrics of interest, such as LERF and CCFP. However, the population dose figures are for all EPRI accident classes.

Each table has three major columns. The first provides the EPRI Accident Class. The second and third provide the results for the base case (ILRT frequency of three per 10 years) and the ILRT frequency of one per 15 years. Columns 2 and 3 are further subdivided to provide the results for the base case (without corrosion [that is, without the potential for age-related corrosion of non-inspectable areas of the containment]), and with corrosion. .Each table contains rows that provide the frequency results for EPRI Accident Classes 3a, 3b, and population dose rates. Additional rows provide the change in dose rates, total and change in conditional containment failure probability (CCFP), and change in LERF. On this table, all delta or changes in values are calculated from the base case of ILRT frequency of three per 10 years.

From inspection of the results for VEGP, the maximum risk change is from the sensitivity case that considers the potential for age-related corrosion of non-inspectable areas of the containment. In this case, the change in CCFP is 0.91%, the change in LERF is 1.44E-07 per year, and the population dose increase is 4.3E-3 person-rem\yr or 11.1% of the base. The total LERF for VEGP, including external events, is estimated to be 4.91E-07 per year and is significantly lower than the threshold for total LERF contained in Regulatory Guide 1.174. It should be noted that while on a percentage basis the change in population dose rates appears high, the total population dose remains very small - 4.32E-02 person-rem/yr at an ILRT frequency of one per 15 years and a change of 4.3E-03 from the base case. These changes in dose and CCFP are within the small range as discussed in Section 2.2.

Table 6-1
Summary of VEGP ILRT Interval Extension Risk Metrics

Risk Metric	Base Case ILRT Frequency (3 per 10 years)		Proposed ILRT Frequency (1 per 15 years)	
	Without Corrosion	With Corrosion	Without Corrosion	With Corrosion
Class 3a Frequency (per year)	1.42E-07	1.42E-07	7.11E-07	7.11E-07
Class 3b Frequency (per year)	3.55E-08	3.57E-08	1.78E-07	1.80E-07
Population Dose Rate (person-rem/yr)	3.84E-02	3.89E-02	4.31E-02	4.32E-02
Change in Dose Rate, person-rem\yr (% of TI base)	N/A		4.26E-03 (11.0%)	4.30E-03 (11.1%)
CCFP	0.79%	0.79%	1.69%	1.70%
Delta CCFP	N/A		0.89%	0.91%
Delta LERF	N/A		1.42E-07	1.44E-07

Table 6-2 provides a summary of the important risk metrics for the CGS ILRT interval extension risk analysis. The risk metric changes are presented for the base and sensitivity cases performed.

From inspection of the results, the maximum risk change is from the sensitivity case that considers corrosion. In this case, the change in CCFP is 0.6%, the change in LERF is 4.58E-08 per year, and the population dose increase is 3.4E-03 person-rem/yr or 0.26%. These increases in CCFP and dose are within the small range discussed in Section 2.2. While not required in this assessment, the total LERF is 3.7E-06 per year.

Table 6-2
Summary of CGS ILRT Interval Risk Metrics

Risk Metric	Base Case ILRT Frequency (3 per 10 years)		Proposed ILRT Frequency (1 per 15 years)	
	Without Corrosion	With Corrosion	Without Corrosion	With Corrosion
Class 3a Frequency (per year)	4.05E-8	4.05E-8	2.02E-07	2.02E-07
Class 3b Frequency (per year)	1.01E-08	1.05E-08	5.06E-08	5.59E-08
Population Dose Rate (person-rem/yr)	1.34	1.34	1.34	1.34
Change in Dose Rate, person-rem\yr (% of Tl base)	N/A		3.15 E-3 (0.24)	3.43E-3 (0.26)
CCFP	70.1%	70.1%	70.7%	70.8%
Delta CCFP	N/A		0.6%	0.6%
Delta LERF	N/A		4.05E-08	4.58E-08

6.2 Conclusions

This analysis confirms the findings of earlier studies: reducing the frequency of Type A tests (ILRTs) from the current three per 10 years to one per 15 years leads to a small increase in risk.

Using the conservative assumptions concerning the leakage and timing associated with a large early release, the reduction in frequency of the type A ILRT test results in a change in LERF that ranges between the “very small” (< 1E-07) and “small” (1E-07 to 1E-06) risk increase regions of Regulatory Guide 1.174. In the cases where the risk increase is conservatively calculated to be greater than the “very small” region, the total LERF is significantly lower than the Regulatory Guide 1.174 threshold guideline of total LERF less than 1E-05 per year. The core damage frequency remains unchanged.

Other figures-of-merit have similar small changes, including the population dose rate and the conditional containment failure probability (CCFP), which change very little over the range of ILRT frequency from three in 10 years to one in 15 years. Acceptance guidelines for dose and CCFP are discussed in Section 2.2.

The following table summarizes figures-of-merit information for 59 plants that have applied for one-time ILRT interval extensions, from Table G-1:

Table 6-3
Summary of Figures-of-Merit Information from Plants That Applied for One-Time ILRT Interval Extensions

	Δ LERF	Δ CCFP %	Population dose change, person-rem/yr[9]	Population dose change, %
Minimum	1E-9	0.003	<0.01	.002
Maximum	7.6E-7	1.1	0.2	0.46

Defense-in-depth as well as safety margins are maintained through the continued inspection of containment as required by ASME Section XI, Subsections IWE and IWL, and other required inspections, such as those performed to satisfy the Maintenance Rule. In addition, NEI 94-01 [16] requires acceptable historical performance of Type A Integrated Leak Rate Tests before integrated leak rate testing intervals can be extended.

As can be seen from the two examples in this report as well as the analyses developed to date for 59 one-time interval extensions, the results of the analyses, and therefore the conclusions derived from them, are applicable to a large number of plants.

Given the above, the risk impact associated with the extension of ILRT frequency from three per 10 years to one per 15 years is small and is generally applicable, and it could well be generically applicable to the current fleet of operating nuclear units. However, because of the possibility of an outlying plant, a confirmatory risk impact assessment is prudent to provide plant-specific assurance of the acceptability of the risk impact of extending ILRT intervals up to a maximum of 15 years.

[9] As noted in Section 2.2, the one-time extensions assumed a large leak (EPRI class 3b) magnitude of 35La, and this analysis uses 100La. This change would have resulted in higher incremental dose increases.

7 REFERENCES

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A
ILRT DATA

This appendix provides the database of ILRT events. These events are taken from two NEI utility surveys [8, 9], NUREG-1493, and other events from industry data, such as LER and reportable event reports. A summary of the data is provided in Table A-1.

Also in Table A-1 are the ILRT events that have occurred after the original version of this report (Revision 0). These events are included in Table A-1 as numbers 72 through 75.

The following provides a summary of the columns of Table A-1:

- Column 1 provides a numerical entry number.
- Column 2 provides the date of the event. This date is either the date of the actual occurrence of the event or the date associated with the reporting of the event.
- Column 3 provides the plant unit name and containment type. For some older ILRT data from the NUMARC survey it was not possible to attribute the data to a unit or containment type. In these cases, the NUMARC survey reference number is provided in this column.
- Column 4 provides a reference to the source of the data.
- Column 5 provides the resulting leakage in fraction of La. Where additional information on the type of leakage is available (such as the fraction due either to an “A” type test or to “B&C” type tests), this information is provided in this column.
- Column 6 provides the leakage rate in either standard cubic centimeters per minute (SCCM) or percent per day where this information is available from the data.
- Column 7 provides the method of detection. The methods of detection include ILRT, LLRT Penalty, Verification Test, observation, or inspection.
- Column 8 provides a brief summary of the cause of the failure.
- Column 9 provides a brief description of the event.

Columns 10 through 16 are used to sort the ILRT event data for support of the expert elicitation. Many of the entries are duplicate information to previous columns with less detail (i.e., information was categorized) for easy sorting. The columns are retained since they provide a documentation record of the information provided to expert elicitation.

- Column 10 provides the method of detection note. Detection methods are as follows:

A	Type A test
B	Type B/C local test
I	Invalid
L	Low pressure monitoring
O	Operator inspection/other inspection
V	Visual exam/inspection

- Column 11 provides an initial screening on whether an extended ILRT interval would impact the time that the event would go undetected.
- Column 12 provides a cause category or failure mode. Cause categories range from 1 to 10 as follows:

1	Original Containment Design Deficiency
2	Construction Error or Deficiency (for example, construction debris in concrete)
3	Human error associated with testing or maintenance (for example, not replacing instrument caps, mechanical misalignment)
4	Human error, design error, or other deficiency associated with design or modifications (for example, spare pipes not capped, debris left in system)
5	Erosion
6	Corrosion
7	Fatigue failures
8	Unknown or other
9	Testing and/or procedural errors
10	ILRT exceeded due to B &/or C leakage penalty or SG manway gasket leak

- Column 13 provides the leakage size category. Leakage size categories are as follows:

S	Small, <2 La
M	Medium, 2 - 10 La
L	Large, >10 La
U	Unknown
N/A	No excessive leak

- Column 14 provides the detection category. Detection category is assigned as ILRT or Other.
- Column 15 provides the containment size applicability.
- Column 16 provides a notes column. A list of the table notes is provided following the table.

Table A-1
Tabulation and Characterization of Historical ILRT Events

No.	Date	Unit or Reference Contmt Type	Reference LER, Report, and so on	Leakage, Fraction of La	La, SCCM or %/day	How Detected	Cause	Description	Detection Method	Will ILRT Interval Affect Non Detection Time?	Cause Category (Failure Mode)	Size of Leakage	Detected by	Containment Size Applicability	Notes
1	Mar-77	NUMARC note unknown	NUMARC letter 2/18/94 to NRC	Unknown	Unknown	ILRT	Holes inadvertently drilled in liner		A	Yes	4	U	ILRT	All	
2	Apr-77	NUMARC 24 PWR	NUMARC 24 PWR	Unknown, >1La	175000	ILRT	SG manway gasket leak	Excessive leakage identified by ILRT	A, O	No	10	U	ILRT	None	1
3	Mar-78	NUMARC 4 PWR	NUMARC 4 PWR	0 .88 La+ (B&C)	346800	ILRT	SG manway gasket leak	Excessive leakage identified by ILRT	A, O	No	10	S	ILRT	None	1
4	Jun-80	NUMARC 25 unknown	NUMARC 25 unknown	0.072 La+ (B+C)	538000	LLRT Penalty		Excessive C local leakage identified by LLRT	B, A	No	10	S	Other	None	
5	Feb-81	NUMARC 21 unknown	NUMARC 21 unknown	N/A		Verification test		ILRT Exceedance due to instrument verification test discrepancy	A, I	No	9	N/A	ILRT	None	2
6	Jun-82	NUMARC 4, unknown	NUMARC 4 unknown	0.43 La+ (B&C)	346000	ILRT	Lineup Error	ILRT Exceedance due to lineup error. Not real leakage.	A, I	No	9	N/A	ILRT	None	2
7	Aug-83	NUMARC 19 unknown	NUMARC 19 unknown	1.3La	83200	LLRT		Excessive local leakage identified by LLRT	B, A	No	10	S	Other	None	
8	Apr-84	NUMARC 25 unknown	NUMARC 25 unknown	0.031 La+ (B&C)	538000	LLRT penalty		Excessive local leakage identified by LLRT	B, A	No	10	S	Other	None	
9	Aug-84	NUMARC 28 unknown	NUMARC 28 unknown	0.071 La(A) 14.91 La w/(B&C)	95330	LLRT penalty		Excessive C local leakage identified by LLRT	B, A	No	10	L	Other	None	
10	Jun-85	NUMARC 26 unknown	NUMARC 26 unknown	0.19 La(A) 20.82 La w/(B&C)	862307	LLRT penalty		Excessive B&C local leakage identified by LLRT	B, A	No	10	L	Other	None	
11	Nov-85	NUMARC 3 unknown	NUMARC 3 unknown	0.36 La (A) 1.89 La w/(B&C)	211600	LLRT penalty	`	Excessive C local leakage identified by LLRT	B, A	No	10	S	Other	None	
12	Apr-86	NUMARC 28 unknown	NUMARC 28 unknown	<0.05 La(A) <9.55 La w/(B&C)	95330	LLRT penalty		Excessive C local leakage identified by LLRT	B, A	No	10	M	Other	None	
13	May-86	NUMARC 23 unknown	NUMARC 23 unknown	0.27 La(A) 0.99 La w/(B&C)	135920	LLRT penalty		Excessive B&C local leakage identified by LLRT	B, A	No	10	S	Other	None	

Table A-1 (continued)
Tabulation and Characterization of Historical ILRT Events

No.	Date	Unit or Reference Contmt Type	Reference LER, Report, and so on	Leakage, Fraction of La	La, SCCM or %/day	How Detected	Cause	Description	Detection Method	Will ILRT Interval Affect Non Detection Time?	Cause Category (Failure Mode)	Size of Leakage	Detected by	Containment Size Applicability	Notes
14	Jun-86	Susquehanna 2 BWR Mark 2	NUREG-1493	2.6 La	1.00%	ILRT		ILRT prior to LLRT	A, B	No	8	M	ILRT	None	5a
15	Nov-86	Quad cities-2 BWR Mark 1	NUREG-1493	0.88 La	1.00%	ILRT	Faulty drywell head gasket	Excessive local leakage identified by ILRT	A, B	No	3	S	ILRT	All	7a
16	Nov-86	TMI-1 PWR Large dry	NUREG-1493	1.0 La	0.10%	ILRT		ILRT prior to LLRT	A, B	No	8	S	ILRT	None	5a
17	Nov-86	NUMARC 24 PWR	NUMARC 24 PWR	1.0 La, 1.0 La w/(B&C)	175000	ILRT	SG manway gasket leak	Excessive leakage identified by ILRT	A, O	No	10	S	ILRT	None	1
18	Aug-87	NUMARC 27 PWR	NUMARC 27 PWR	0.027 La (A) 2.46 La w/(B&C)	236203	LLRT penalty		Excessive local leakage identified by LLRT	B, A	No	10	M	Other	None	
19	Sep-87	Quad cities-1 BWR Mark 1	NUREG-1493	Unknown		ILRT		ILRT without prior LLRT	A, B	No	8	U	ILRT	None	5a
20	Sep-87	NUMARC 28 unknown	NUMARC 28 unknown	0.43 La + (B&C)	287407	LLRT penalty		Excessive B&C local leakage identified by LLRT	B, A	No	10	S	Other	None	
21	Sep-88	NUMARC 30 unknown	NUMARC 30 unknown	Unknown	218503	LLRT penalty		Excessive C local leakage identified by LLRT	B, A	No	10	U	Other	None	
22	Oct-89	Harris-1 PWR large dry	NUREG-1493	Unknown		ILRT		ILRT without prior LLRT, as found not quantified	A, B	No	8	U	ILRT	None	5a
23	Nov-89	Hatch-2 BWR Mark 1	NUREG-1493	0.86 La	1.20%	LLRT penalty		Excessive local leakage identified by LLRT	B, A	No	10	S	Other	None	
24	Nov-89	Fermi-2 BWR Mark 1	NUREG-1493	1.9 La	0.50%	LLRT penalty		Excessive local leakage identified by LLRT	B, A	No	10	S	Other	None	
25	Dec-89	Beaver Valley-1 PWR Subatm	NUREG-1493	Unknown	0.10%	ILRT	Two penetration leaks discovered during ILRT	Excessive local leakage identified by ILRT and not identified by LLRT	A	Yes, however ILRT interval would not be extended under NEI 9401	3	U	ILRT	All	8
26	Feb-90	Dresden 3 BWR Mark 1	NUREG-1493	0.78 La	1.60%	LLRT penalty		Excessive local leakage identified by LLRT	B, A	No	10	S	Other	None	

Table A-1 (continued)
Tabulation and Characterization of Historical ILRT Events

No.	Date	Unit or Reference Contmt Type	Reference LER, Report, and so on	Leakage, Fraction of La	La, SCCM or %/day	How Detected	Cause	Description	Detection Method	Will ILRT Interval Affect Non Detection Time?	Cause Category (Failure Mode)	Size of Leakage	Detected by	Containment Size Applicability	Notes
27	Feb-90	Brunswick-2 BWR Mark 1	NUREG-1493	0.94 La	0.50%	LLRT penalty		Excessive local leakage identified by LLRT	B, A	No	10	S	Other	None	
28	May-90	Sequoyah-1 PWR ice condenser	NUREG-1493	2.8 La	0.25%	LLRT penalty		Excessive local leakage identified by LLRT	B, A	No	10	M	Other	None	
29	May-90	Sequoyah-2 PWR ice condenser	NUREG-1493	<1.0 La	0.25La	ILRT	Penetration leakage, faulty LLRT	Excessive local leakage identified by ILRT and not identified by LLRT	A, B	No	8	S	ILRT	None, leakage <La	5b
30	Jun-90	LaSalle-2 BWR Mark 2	NUREG-1493	Unknown, >La	0.63%	Unknown			A	Maybe	8	U	ILRT	None	
31	Jun-90	Trojan PWR large dry	NUREG-1493	Unknown	1.30%	ILRT	Instrumentation problems		A, I	No	9	N/A	ILRT	None	2
32	Sep-90	NUMARC 31, unknown	NUMARC 31 unknown	Unknown	218503	LLRT penalty		Excessive C local leakage identified by LLRT	B, A	No	10	U	Other	None	
33	Oct-90	Callaway PWR large dry	NUREG-1493	Unknown, >La	0.20%	ILRT	Penetration leakage	Excessive local leakage identified by ILRT	A	Yes, however ILRT interval would not be extended under NEI 9401	3	U	ILRT	All	8
34	Oct-90	NUMARC 20 unknown	NUMARC 20 unknown	1.7 La w/(B&C)	188945	ILRT		Excessive B&C local leakage identified by ILRT and not identified by LLRT	A, B	Maybe	4	S	ILRT	All	5a
35	Dec-90	Dresden 2 BWR Mark 1	NUREG-1493	15.3 La	1.60%	ILRT	Vacuum breaker leakage discovered during ILRT	Excessive local leakage identified by ILRT	A, B	Maybe	3	L	ILRT	All	5a
36	Feb-91	Braidwood 1 PWR large dry	NUREG-1493	0.56 La	0.10%	ILRT	Type B failure found during ILRT w/outer doors open, airlock hatch shaft seal	Excessive local leakage identified by ILRT and not identified by LLRT	A, B	Maybe	3	S	ILRT	None, leakage <La	5a
37	Feb-91	Brunswick 1 BWR Mark 1	NUREG-1493	0.99 La	0.50%	LLRT penalty		Excessive local leakage identified by LLRT	B, A	No	10	S	Other	None	

Table A-1 (continued)
Tabulation and Characterization of Historical ILRT Events

No.	Date	Unit or Reference Contmt Type	Reference LER, Report, and so on	Leakage, Fraction of La	La, SCCM or %/day	How Detected	Cause	Description	Detection Method	Will ILRT Interval Affect Non Detection Time?	Cause Category (Failure Mode)	Size of Leakage	Detected by	Containment Size Applicability	Notes
38	Apr-91	NUMARC 2 unknown	NUMARC 2 unknown	0.47 La (A) 0.84 La w/(B&C)	163000	ILRT		Excessive B&C local leakage identified by ILRT and not identified by LLRT	A, B	Maybe	8	S	ILRT	None, leakage <La	5a
39	Jun-91	Millstone-1	NUREG-1493	Unknown, >0.75 La	1.20%	LLRT penalty		Excessive local leakage identified by LLRT	B, A	No	10	U	Other	None	
40	Jun-91	BWR		0.29 La+ (B&C)	236203	LLRT penalty		Excessive C local leakage identified by LLRT	B, A	No	10	S	Other	None	
41	Jul-91	Pilgrim BWR Mark 1	NUREG-1493, LER 91-023-00	1.2 La	1.00%	ILRT	Drywell head bolts loose, improper spherical washer material	Failure of spherical washers led to loosening of 11 of 76 bolts, drywell head contribution .74%/day	A, O, B	Probably not	4	S	ILRT	Small	7b
42	Sep-91	Braidwood 2 PWR large dry	NUREG-1493	0.55 La	0.10%	ILRT	Several local leaks found during ILRT w/outer doors open	Excessive local leakage identified by ILRT and not identified by LLRT	A, B	Maybe	8	S	ILRT	None, leakage <La	5a
43	Dec-91	Brunswick 2 BWR Mark 1	NUREG-1493	0.79 La	0.50%	LLRT penalty		Excessive local leakage identified by LLRT	B, A	No	10	S	Other	None	
44	Dec-91	PVNGS-2 PWR large dry	NUREG-1493	0.83 La	0.10%	LLRT penalty		Excessive local leakage identified by LLRT	B, A	No	10	S	Other	None	
45	Dec-91	Cooper BWR Mark 1	NUREG-1493, LER 91-020-00	1.4 La	149623	ILRT	Structural failure of radiation monitor	Radiation monitor breached its shield chamber during ILRT pressurization at 51 psig. Leakage from monitor path =0.61 La.	A	Yes, not a pre-existing leak	4	S	ILRT	All	
46	Mar-92	Dresden-3 BWR Mark 1	NUREG-1493	Unknown, >La	1.60%	LLRT penalty		Excessive local leakage identified by LLRT	B, A	No	10	U	Other	None	
47	Mar-92	LaSalle-2 BWR Mark 2	NUREG-1493	0.56 La	0.63%	LLRT penalty		Excessive local leakage identified by LLRT	B, A	No	10	S	Other	None	

Table A-1 (continued)
Tabulation and Characterization of Historical ILRT Events

No.	Date	Unit or Reference Contmt Type	Reference LER, Report, and so on	Leakage, Fraction of La	La, SCCM or %/day	How Detected	Cause	Description	Detection Method	Will ILRT Interval Affect Non Detection Time?	Cause Category (Failure Mode)	Size of Leakage	Detected by	Containment Size Applicability	Notes
48	Apr-92	Sequoyah-2 PWR ice condenser	NUREG-1493	1.68 La	0.25%	LLRT penalty		Excessive local leakage identified by LLRT	B, A	No	10	S	Other	None	9
49	Apr-92	Vogtle-2 PWR large dry	NUREG-1493, NUMARC 1	0.62 La (A) >.75 La w/(B&C), Unknown	360000 0.2%	LLRT penalty		Excessive B&C local leakage identified by LLRT	B, A	No	10	U	Other	None	9
50	May-92	ANO-1 PWR large dry	NUREG-1493	Unknown, >La	0.20%	LLRT penalty		Excessive local leakage identified by LLRT	B, A	No	10	U	Other	None	9
51	Aug-92	River Bend BWR Mark 3	NUREG-1493	Unknown, >La	0.26%	LLRT penalty		Excessive local leakage identified by LLRT	B, A	No	10	U	Other	None	
52	Sep-92	NUMARC 21 PWR		1.3 La+ (B&C)	442525	ILRT	SG manway gasket leak	Excessive leakage identified by ILRT	A, O	No	10	S	ILRT	None	1
53	Oct-92	Fermi-2 BWR Mark 1	NUREG-1493	< 2 La	0.50%	LLRT penalty		Excessive local leakage identified by LLRT	B, A	No	10	S	Other	None	
54	Nov-92	Hatch-2 BWR Mark 1	NUREG-1493	1.11 La	1.20%	LLRT penalty		Excessive local leakage identified by LLRT	B, A	No	10	S	Other	None	
55	Nov-93	NUMARC 3 Unknown	NUMARC 3 unknown	0.21 La(A) 1.34 La w/(B&C)	211600	ILRT	Lineup error	Excessive local leakage identified by ILRT due to lineup error	A, I	No	9	N/A	ILRT	None	2
56	Feb-94	Ginna PWR large dry	LER 94-003-00	Unknown		I&C observation	Instrument plug not installed	Instrument plug not installed following I&C work. Procedures enhanced to ensure installation in future	O	No	3	N/A	Other	None	3
57	Feb-94	Surry 1 PWR Subatm	LER 94-003-00	>La		Piping inspection	Failure of coal tar epoxy coating followed by corrosion	Hole in piping for recirculation spray water heat exchanger	V, A	No	6	U	Other	All	4
58	Mar-94	Braidwood 1 PWR large dry	LER 94-003	0.9 La	216908 0.1%	ILRT	Construction deficiency not previously identified	Concrete vent pipes associated with emergency hatch not capped. Leakage from vent pipes = 0.09 La.	A	Yes	4	S	ILRT	None, leakage <La	

Table A-1 (continued)
Tabulation and Characterization of Historical ILRT Events

No.	Date	Unit or Reference Contmt Type	Reference LER, Report, and so on	Leakage, Fraction of La	La, SCCM or %/day	How Detected	Cause	Description	Detection Method	Will ILRT Interval Affect Non Detection Time?	Cause Category (Failure Mode)	Size of Leakage	Detected by	Containment Size Applicability	Notes
59	Apr-94	Sequoyah-1 PWR ice condenser	LER 94-005-00	0.75-1.0 La	0.25%	Inability to maintain PRT P	Circumferential crack in RV bellows	This bellows failure was detected during normal operation	O, A	No	7	S	Other	All	
60	Dec-94	Pilgrim BWR Mark 1	LER 94-007-00	>La	1.00%	I&C inspection	Instrument plug not installed	Plug for torus-atmosphere dp transmitter not installed; corrective action includes verification surveillance	O, A	No	3	U	Other	None	10
61	Apr-95	Vermont Yankee BWR Mark 1	NEI Survey	2 La	0.80%	ILRT	Excessive local leakage	Valves contaminated with construction debris after passing LLRT	A, B	Maybe	4	M	ILRT	All	5a
62	Sep-95	Indian Point 3 PWR Large Dry	LER 95-019-00	Insignificant leakage	0.10%	Inspection/ radiograph	Excessive local leakage	Through-wall cracks on pipe caps on spare penetration due to contaminated stagnant water	V, O	No	6	N/A	Other	None	6
63	Feb-96	Surry 2 PWR Subatm	LER 96001	Unknown		Observation at power		Leaking weld on return pipe from refueling cavity to RWST	O, A	No	6	N/A	Other	All	
64	Oct-96	Oyster Creek BWR Mark 1	LER 96-011-0	2 La		Low pressure monitoring	Vacuum breaker valve cover leaking	Misalignment of valve cover during assembly, shifting during heatup	L, A	No	3	M	Other	All	5a
65	Sep-99	North Anna 2 PWR Subatm	NEI Survey, LER 1999-002-00	0.07 La		Liner coating inspection	1/4" defect hole	Wooden timber in concrete in back of liner. Leakage through defect = 0.07 La.	V, A	No	2	S	Other	All	
66	Nov-99	PVNGS 1 PWR large dry	LER 2000-004	Insignificant leakage	0.10%	ILRT	Inadequate procedure for LLRT of Purge valves, valve seat adjustment	Purge valve penetration leakage identified during ILRT.	B	No	3	N/A	ILRT	None	2

Table A-1 (continued)
Tabulation and Characterization of Historical ILRT Events

No.	Date	Unit or Reference Contmt Type	Reference LER, Report, and so on	Leakage, Fraction of La	La, SCCM or %/day	How Detected	Cause	Description	Detection Method	Will ILRT Interval Affect Non Detection Time?	Cause Category (Failure Mode)	Size of Leakage	Detected by	Containment Size Applicability	Notes
67	Nov. 99, Jan. 00	Cook 2 PWR Ice Condenser	NEI Survey. LER 2000-001-01	Unknown < La		Liner, coatings inspection	Inadequate repair of hole drilled during construction.	Coatings and ISI inspection discovered rusting area. Further investigation dislodged repair material and a 3/16" hole. Prior ILRT (1992) successful.	V	No	2	N/A	Other	All	
68	99	Brunswick 2 BWR Mark 1	NEI survey	Unknown < La	0.50%	IWE inspection	Three through-wall defects in liner	Pitting corrosion and debris in concrete	V, A	No	2	S	Other	All	
69	Aug-01	PVNGS-3 PWR large dry	Non-emergency event report 8/17/01	Unknown	0.10%	Operations monitoring containment sump	Quick opening closure device not properly closed, or loosening of device in service.	Fuel transfer tube quick operating closure device leak path.	O, A	No	3	U	Other	None	5a
70	Oct-01	Vermont Yankee BWR Mark 1	Non-emergency event report 10/30/2001	Unknown > La	0.80%	Operator observation and isolation		Tube broke on discharge of H2O2 monitor sample pump.	O, A	No	4	U	Other	All	11
71		Vermont Yankee BWR Mark 1	NUREG-1493	1.0 La	0.80%	ILRT	Drywell manway penetration leakage		A, B	Maybe	3	S	ILRT	Small	5a
72	May-05	Vermont Yankee	LER 2001-02-01	>.6La	0.80%	LLRT	Design of internal nozzle check valve components	Local leak rate through two series turbine exhaust line nozzle check valves excessive	B	No	4	>.6La	Other	None	
73	Dec-02	Palisades	LER 2002-003-00	Unknown	Unknown	Technician Observation	Human performance inadequacy; pipe caps not labeled or on check list	Test caps not replaced during restoration from ILRT	V	No	3	Unknown	Other	All	
74	Nov-04	Cook-2	LER 2004-004-00	>0.6La	Unknown	Operator Observation	Failure to have a positive comprehensive program for containment integrity or closure components	Two small drain valves left open for ~1 week	Operator noted lack of normal containment venting	No	3	>0.6La	Other	All	

Table A-1 (continued)
Tabulation and Characterization of Historical ILRT Events

No.	Date	Unit or Reference Contmt Type	Reference LER, Report, and so on	Leakage, Fraction of La	La, SCCM or %/day	How Detected	Cause	Description	Detection Method	Will ILRT Interval Affect Non Detection Time?	Cause Category (Failure Mode)	Size of Leakage	Detected by	Containment Size Applicability	Notes
75	Jun-05	Fitzpatrick	50-72 Report No. 41815, LER 05-003	As-found leakage very small, 1-2 drops per minute water.	0.5%/day	Observed leakage during inspection and walkdown for activities not related to containment inspection or leak rate testing.	HPCI turbine exhaust condensation oscillation pressure pulses causing high cycle fatigue failure. Crack propagation driven by HPCI operation. Design of HPCI turbine exhaust did not include a sparger.	Slight weepage (1-2 drops/min.) noted during inspection walkdown outside torus. An "X" shaped crack with a maximum length of 4.6" discovered below water line.	O	No	7	S	Other	None	12

1 Steam generator manway leakage is detectable during startup and normal operation.
Monitored via Technical Specification identified and unidentified leakage limits.

2 The event does not appear to be the result of ILRT failure or true containment leakage.

3 Leakage pathway from containment to atmosphere would exist only when the equipment hatch inner door was open.

4 Radiation monitors and isolation valves are also provided. Fluid leakage would be detected by subsequent piping inspections.

5a Leakage pathway would be identified in next local leak rate test (LLRT).

5b This leakage path should have been identified by LLRT. Would be discovered during subsequent LLRTs, after correction of faulty LLRT.

6 Containment integrity was not an issue as the penetration was pressurized and monitored.

7a Leaking drywell head gasket would have been replaced at next refueling.

7b Had this not been identified in an ILRT, loose bolts and washer failures should have been identified and replaced in the next refueling.

8 If leakage cannot be identified by local testing, Type A test does not meet NEI 94-01 performance criteria for ILRT interval extension.

9 ILRT La exceeded due to B&C Leakage Penalty Identified by LLRT.

10 This pathway would probably have been identified in the next instrument calibration cycle.

11 Engineering evaluation determined that under accident conditions, leakage would have exceeded allowable leakage limits.

12 This leak path may not have been detected in the course of an ILRT by measuring containment leakage rate due to its location under the torus water level and because of its size. An engineering evaluation determined the crack size to be less than critical and would have remained small during the HPCI emergency mission time. The crack was calculated to grow but remain stable below the critical crack length during HPCI operation with containment pressurized.

B
EXPERT ELICITATION PROCESS

This appendix provides an overview of the expert elicitation process [11, 12] and its application to the solicitation of expert opinion for the ILRT Type A Testing Interval Optimization Project. The process is based on the “Recommendations for Probabilistic Seismic Hazard Analysis: Guidance on Uncertainty and Use of Experts” (NUREG/CR-6372 [12]) and “Branch Technical Position on the Use of Expert Elicitation in the High-Level Radioactive Waste Program” (NUREG-1563 [11]).

B.1 Introduction to the Elicitation Process

The goal of the expert elicitation process is to obtain frequency and magnitude estimates for containment leakage that would not be detected by other inspections, tests, or alternative means.

There are five functional requirements of the expert elicitation process. These five requirements are:

- Requirement 1: Identification of the expert judgment process
- Requirement 2: Identification and selection of experts
- Requirement 3: Determination of the need for outside expert judgment
- Requirement 4: Utilization of either the technical integrator (TI) or technical facilitator/integrator (TFI) process
- Requirement 5: Responsibility for the expert judgment

The five functional requirements of the expert judgment process identify the issue, identify the experts, outline the process used in the solicitation of their opinion, and specify the use of their judgment in the ILRT Type A testing interval optimization process. Each of the five functional requirements is discussed in detail in the following sections.

B.2 Expert Elicitation Summary

The goal of the expert elicitation process is to determine the probability and magnitude of containment leakage events. The probability and magnitude of containment leakage events will be used in the determination of the risk impact associated with the ILRT Type A testing interval optimization.

The expert elicitation process inputs are derived from an ILRT events database consisting of information collected via NEI surveys, LERs, and NRC reports (NUREG-1493). The expert elicitation process uses a facilitated expert meeting that considers data, containment design, maintenance, and testing. The process was consistent with the approach described in References [11] and [12].

Using the process outlined in those two references, the ILRT Type A testing interval optimization has been assigned a degree of importance of Degree II and a level of complexity of C. These assignments indicate that a TI process is sufficient for the expert panel process. In the case of a level of complexity of Level C, a facilitated expert panel meeting is required to solicit the opinions of the technical community. Through a nomination process, experts are selected. Each of the experts has significant expertise in areas related to containment structures and/or containment testing.

The technical integrator facilitates the expert panel meeting in which the problem statement is provided. The problem statement includes an ILRT events database and potential approaches (in addition to expert elicitation) and their results. The expert panel then provides its individual judgments. The technical integrator integrates the individual results to obtain the community distribution. The community distribution is provided to the expert panel to ensure agreement with the final community distribution. The results are then used in the risk impact assessment.

B.3 Requirement 1: Identification of the Expert Judgment Process

There are several forms that the expert elicitation process can take depending on the complexity of the issue, the resources available to address the issue, and other factors. This requirement provides the outline of the expert judgment process based on these factors. Three topics are discussed in the following report sections that assist in the determination of the details of the expert elicitation process. These topics are:

- Defining the specific issue
- Determining the degree of importance and degree of complexity of the issue
- Deciding whether to use a TI or TFI

B.3.1 Defining the Specific Issue

The technical issue for which expert judgment is to be applied needs to be defined clearly and narrowly enough that it is possible to identify the relevant expertise and to use it correctly. Defining the technical issue requires the following tasks:

- Clearly identify the issue such that one or more technical experts can be selected.
- Define how the issue fits into the PRA.
- Allow the experts to redefine the issue that allows the experts to provide input.

The issue associated with the optimization of ILRT Type A testing interval has been clearly defined in the ILRT problem statement. Therefore, this requirement is considered satisfied.

B.3.2 Determining the Degree of Importance and Level of Complexity

In the following sections, the process used to determine the degree of importance and level of complexity of the ILRT testing optimization is discussed.

B.3.2.1 Determining the Degree of Importance

To assist the experts in the expert elicitation process as well as to define the form of the process, it is necessary to classify the technical issue into one of three degrees. These three degrees, defined as Degree I, Degree II, and Degree III, are intended for use in the determination of the expert elicitation process to be used. The determination of the degree of importance is based on technical criteria only. The degree characterizations are as follows:

- Degree I: Non-controversial issue and/or not significant to the overall results of the analysis.
- Degree II: Issue has significant uncertainty or diversity of opinion; controversial; moderately significant to the overall result of the analysis; and/or moderately complex.
- Degree III: Highly contentious issue; very significant to the overall result of the analysis; and/or highly complex.

In assigning the degree of importance to an issue, there is some judgment necessary because the degree categories represent a coarse partition of the range of potential degrees.

In the case of the optimization of the ILRT testing intervals, Degree II is selected. Degree I is not chosen because the results of the expert elicitation process are indeed significant to the results of the analysis. In fact, a case could be made that the results of the expert elicitation process are very significant to the results of the analysis, necessitating an assignment of a Degree III. However, the sensitivity of the results of the analysis to the expert elicitation process is mitigated by the availability of significant amounts of data. These data, although not complete enough to perform the analysis, do provide information upon which the experts can base their judgments. In addition, experts will be chosen for the knowledge of the mechanisms that can result in containment leakage events and therefore provide additional assurance that their judgment is only moderately significant to the overall result. Last, the issue of testing extension and specifically ILRT Type A test optimization is not considered highly complex, nor is the issue considered highly contentious. Therefore, the assignment of degree of importance of Degree II is appropriate.

B.3.2.2 Determining the Level of Complexity

Once the degree of the issue has been selected, it is necessary to select the level of complexity. There are four levels of complexity, defined as Level A, B, C, and D. A key input to the assignment of the level of complexity is the degree of importance. The degree of importance captures how complex and how controversial the issue is, but alone is not sufficient for the choice of the level of complexity.

In summary, levels of complexity of A, B, or C are characterized by the TI approach. In the technical integrator approach, the technical integrator plays the role of "evaluator." Input to the technical integrator varies depending on the level of complexity assigned to the issue from basing judgments on his/her own experience and literature to obtaining input through the communication with other experts.

With an issue of a level of complexity of A, the technical integrator's role is to evaluate and weight models based on literature review and experience. With a level of complexity of A, the technical integrator would estimate the community distribution.

With an issue assigned a level of complexity of B, the technical integrator's role is to conduct a literature review and contact those individuals who have developed interpretations or who have particular relevant experience and develop the community distribution.

With an issue assigned a level of complexity of C, the technical integrator's role is to gain additional insight by bringing together experts and focusing their interactions. In the sessions with the technical experts, the experts are given an opportunity to explain their hypotheses, data, and bases. Proponents or advocates of particular technical positions are asked to describe and defend their positions to the other experts. As with levels A and B, the technical integrator develops the community distribution.

Issues assigned a level of complexity of D are characterized by the TFI approach. In level D, a group of expert "evaluators" is identified and their judgments elicited. The technical facilitator/integrator is responsible for identifying the roles of the proponents and evaluators and for ensuring that their interactions provide an opportunity for focused discussion challenge. In the Level D analysis, resources permit and the situation dictates multiple evaluators, and hence a technical facilitator integrator takes responsibility for the aggregated product. The TFI organizes and manages interactions among the proponents and evaluators, identifies and mitigates problems that could develop during the course of the study (for example, an expert who is unwilling or unable to play the evaluator role), and ensures that the evaluators' judgments are properly represented and documented.

Regardless of the level of the study, the goal in the various approaches is the same: to provide the community distribution, which is defined as a representation of the informed technical community's view of the important components and issues and, finally, the result. Also, regardless of the level of the study, a peer review is performed to review the process and substance of the study.

The level of complexity of the ILRT Type A testing optimization is chosen as Level C. The factors affecting this assignment include but are not limited to regulatory issues, public and technical community perception, and resource constraints.

A level of complexity of D is not chosen because empirical data are available that provide an indication of the range of the result of the final analysis. In addition, the phenomena related to containment leakage events are generally understood. In addition, the conceptual models that are involved in the optimization of the ILRT testing interval and potential containment leakage events are relatively limited. Given the required resources and the above discussion, a complexity level of D is not chosen.

Assignment of a level of complexity of A is rejected because it does not significantly involve the technical community in the development of the analysis. Given the regulatory nature of the analysis, it is important to involve the technical community in the development of the analysis.

While a level of complexity of B does involve the technical community, it does not provide a forum for the exchange of alternate conceptual models. Therefore, a level of complexity of B is also not chosen.

A level of complexity of C provides the optimum use of resources because it allows the technical community to participate in the development of the analysis results and the proposal of alternate conceptual models while limiting the resources associated with the solicitation of the expert judgment.

B.4 Requirement 2: Identification and Selection of Experts

One or more evaluators (individuals capable of evaluating the relative credibility of multiple alternative hypotheses to explain the available information) need to be identified. In addition, other experts, such as proponents (experts who advocate a particular hypothesis or technical position), as well as resource experts (technical experts with knowledge of a particular area of importance to an issue) will also be identified and nominated for participation.

Experts will be nominated to the panel by the ILRT optimization project manger. Experts should have extensive nuclear power experience and expertise in one or more of the following areas:

- Containment structure testing and/or maintenance
- Performing ILRTs or interpreting/characterizing ILRT results
- Statistics/probability theory/probabilistic risk assessment
- Failure mechanics

B.5 Requirement 3: Determination of the Need for Outside Expert Judgment

In the case of the ILRT Type A testing optimization, the decision to seek outside (expert elicitation process) expert judgment has already been made, as opposed to using members of the NEI ILRT Optimization Project Team. As previously mentioned, the regulatory nature of the analysis requires that the technical community be involved in the development of the analysis. The selection of the participants will be in accordance with Section 4.4 of this report.

B.6 Requirement 4: Utilize the TI or TFI Process

This requirement is used to determine whether the TI process or the TFI process will be used and to specify the requirements of the process chosen. Because a Level C analysis has been chosen and there is no other basis to decide differently, the TI process is to be used. As described earlier, the TFI process is applied to only Level D analysis. The TI process includes the following significant elements:

- Identifying available information and analysis and information retrieval methods
- Accumulating information relevant to the issue
- Performing the analysis and the data diagnostics
- Developing the community distribution

B.6.1 Identifying Available Information and Analysis and Information Retrieval Methods

The TI is responsible for assembling all relevant technical databases and other information important to the analysis problem at hand, including any data that have been gathered specifically for the analysis. The TI also identifies technical researchers and proponents that he/she intends to contact during the course of the study to gain insight into their positions and interpretations (in a Level C analysis, this means identifying those individuals whom he/she intends to assemble for discussion and interactions). In addition, the TI defines the procedures and methods that will be followed in conducting the analysis.

B.6.2 Accumulating Information Relevant to the Issue, Performing the Analysis, and Developing the Community Distribution

The TI is responsible for understanding the entire spectrum of technical information that is brought to bear on the issue, including written literature, recent works by other experts, and other technical resources. (In advanced technical work, it is always the responsibility of the investigator to learn about the most recent advances in the field, often by direct contact with other experts through personal correspondence, personal meetings, telephone conversations, and so on.) In a level C study, members of the technical community are brought together, and the TI orchestrates interactions and possibly workshops to focus the discussions on the technical issues of most significance to the analysis to be sure that he/she is aware of the diversity in interpretations for these key issues. The TI uses all this information to develop a community distribution of the range of uncertainty for the particular issue being addressed.

B.6.3 Performing the Peer Review

The TI needs to use the peer review team as a sounding board to learn whether the full range of technical views has been identified and assimilated into the project. The ILRT Optimization Project Team will serve as the peer reviewers for the expert panel. In addition, the expert panel will be free to consult other resources as they see necessary.

B.7 Requirement 5: Responsibility for the Expert Judgment

A basic principle is that it is an absolute requirement that there must be a clear definition of the ownership of expert judgments, opinions, and/or interpretations, both as expressed by the individual experts and as integrated together.

In the case of the ILRT Type A testing optimization, assigned a degree of II and a level of complexity of D (see Table B-1), the owner of the process and the results is the technical integrator. The individual experts will own their individual judgments and interpretations.

Table B-1
Degrees of Issues and Levels of Study

Issue Degree	Decision Factor	Study Level
Degree I Non controversial and/or insignificant to the result	Regulatory concern	**Level A** TI evaluates/weighs models based on literature review and experience; estimates community distribution
Degree II Significant uncertainty and diversity; controversial; and complex	Resources available	**Level B** TI interacts with proponents and resource experts to identify issues and interpretations; estimates the community distribution
Degree III Highly contentious; significant to result and highly complex	Public perception	**Level C** TI brings together proponents and resource experts for debate and interaction; TI focuses debate and evaluates alternative interpretations; estimates community distribution
		Level D TFI organizes panel of experts to interpret and evaluate; focused discussions; avoids inappropriate behavior on the part of the evaluators; draws picture of evaluators' estimate of the community's composite distribution; has ultimate responsibility for project

Table B-2
ILRT Expert Elicitation Panel

Name	Experience Summary			
	Degree	**Years Experience**	**Area of Expertise**	**Company, Title, and Selected Experience**
H. Duncan Brewer Panel Member	BS, Nuclear Engineering ME, Mechanical Engineering Registered Professional Engineer	23	Probabilistic risk assessment and safety analysis	Duke Power Company Section manager, severe accident analysis Section manager and lead engineer for nuclear plant probabilistic risk assessment group Lead design engineer responsible for severe accident consequence analysis Integrated nuclear plant safety analysis Chairman, ASME subcommittee on PRA technology
Kenneth Canavan (Facilitator)	BChE, Bachelors of Chemical Engineering Minor in Nuclear Engineering	17	Safety and risk analysis	Data Systems and Solutions Manager, strategic decision support Davis-Besse PRA development Oyster Creek PRA development Three Mile Island PRA development External event PRA development for Oyster Creek and TMI nuclear power stations Lead engineer risk analysis for GPU Decommissioning PRA for Oyster Creek Various risk-informed applications Contributor to peer review process development
John M. Gisclon Panel Member	BS, Mechanical Engineering Registered Professional Engineer	35	Nuclear Power Plant Engineering, Safety Analysis, Testing, and Management	Electric Power Research Institute (EPRI) Nuclear Power Consultant EPRI project manager for risk impact assessment of revised containment leak rate testing intervals (1994) EPRI manager, maintenance technology Developed procedures, conducted and supervised local and integrated leak rate testing at a small BWR and a large PWR

Table B-2 (continued)
ILRT Expert Elicitation Panel

Name	Experience Summary			
	Degree	Years Experience	Area of Expertise	Company, Title, and Selected Experience
Alex McNeill Panel Member	BS, Nuclear Engineering	22	Materials/ Inservice Inspection, IWE/IWL	Dominion Energy Principal Level III Inspector IWE/IWL ISI program administrator Risk-informed inservice inspection program administrator Lead inservice inspection program engineer Member ASME section XI working group on implementation of risk-based examination
James C. Pulsipher Panel Member	BS, Physics MS, Nuclear Engineering	25	Containment Leakage Rate Testing, Containment Systems	U. S. Nuclear Regulatory Commission, Plant Systems Branch, Containment Systems Analyst NRC expert on Appendix J testing Member of ANS 56.8 working group for 19 years Principal NRC participant for revision of 10CFR50, Appendix J, Option B Co-author of Regulatory Guide 1.163 Co-author of recent NRC safety evaluations for one-time extension of ILRT intervals to 15 years
Jim E. Staffiera Panel Member	BS, Mechanical Engineering MBA, Master of Business Administration	32	Containment Fabrication, Erection, and Testing; Containment Inservice Inspection	First Energy Nuclear Operating Company Lead engineer, civil/structural element, design engineering section, nuclear engineering Department, containment inservice inspection program development (ASME subsections IWE/IWL) Chairman, ASME subcommittee (SC) XI working group on containment Member ASME subcommittee (SC) XI Member ASME SC/XI subgroup on water-cooled systems Member ASME SC/XI special working group on editing and review

Table B-2 (continued)
ILRT Expert Elicitation Panel

Name	Experience Summary			
	Degree	Years Experience	Area of Expertise	Company, Title, and Selected Experience
Henry M. Stephens, Jr. Panel Member	BS, Physics and Mathematics	32	Inservice Inspection, NDE	EPRI NDE Center Program manager, NDE training and containment inspection Manager, inservice inspection training NDE training coordinator, NDE instructor Quality assurance engineering Chairman, ASME section XI task group on risk-informed containment inspection Secretary, ASME section XI working group on containment Member, ASME section XI subgroup on water cooled systems

C
EXPERT ELICITATION PREPARATION

This appendix provides a description of the expert elicitation preparation process. Combined with the ILRT problem statement and the ILRT expert elicitation process, this report section provides a full description of the expert elicitation inputs, process, and its application to the risk impact assessment of the ILRT optimization. The ILRT problem statement and the ILRT expert elicitation process are discussed in previous report sections.

The expert elicitation is accomplished in several stages. In the first stage, the experts provide the problem statement. The problem statement contains a statement of issues associated with the extension of the ILRT testing interval as well as information from the Containment Leakage/ Degraded Liner Events database.

In the second stage, the experts are brought together to present the issues as well as the planned approach to the solicitation of their input.

In the third and final stage, the experts are presented with the final results of their collective input ("ILRT failure" probability) as well as the results of the use of their input in the final assessment of the risk impact assessment of the ILRT Type A test interval optimization.

C.1 Stage 1: Expert Elicitation Preparation

In preparation for the expert elicitation meeting, the problem statement as well as the Containment Leakage/Degraded Liner Events database were provided to the experts. As part of the transmittal, experts were requested to provide input to revise the problem statement and focus their collective efforts on the problem. Specifically, experts were asked:

- Does the problem statement adequately address the factors and issues associated with the determination of ILRT failure rate?
- Do you have any suggestions for improvement of the problem statement?
- Was the expert elicitation process adequately described?

In preparation for stage 2, all input received from the experts is incorporated into the problem statements and expert elicitation process.

C.2 Stage 2: Expert Elicitation Meeting

The following section describes the attributes and the detailed agenda of the expert elicitation meeting. The expert elicitation meeting has the following attributes:

- A two-and-one-half-day meeting was planned.
- It is conducted in a location remote to the experts to allow an undistracted ILRT optimization panel meeting.
- The expert elicitation integrator facilitates the meeting.

The planned two-and-one-half-day meeting was organized around the agenda shown in Table C-1.

Table C-1
Expert Elicitation Meeting Agenda

	Day 1 – Morning Session	
	Introductions	8:00–8:30 am
	Presentation of Problem Statement	8:30–9:30 am
	Presentation of the Expert Elicitation Process	9:30–10:00 am
	Break	10:00–10:30 am
	Expert Panel Training	10:30–12:30 pm
	Lunch	12:30–1:30 pm
	Day 1 – Afternoon Session	
	PRA Concepts	1:30–2:30 pm
	Application of PSA Concepts to ILRT Optimization	2:30–3:00 pm
	Break	3:00–3:30 pm
	Presentation of Containment Degradations	3:30–4:30 pm
	ILRT Database and other relevant data	4:30–5:00 pm

Table C-1 (continued)
Expert Elicitation Meeting Agenda

Day 2 – Morning Session		
	Review of Expert Training and ILRT Database	8:00–8:30 am
	Presentation of the Expert Elicitation example	8:30–9:30 am
	Break	9:30–10:00 am
	Expert Discussion of ILRT Issues	10:00–12:00 pm
	Lunch	12:00–1:30 pm
Day 2 – Afternoon Session		
	Expert Discussion of ILRT Issues (continued)	1:30–2:30 pm
	Break	2:30–3:00 pm
	Individual Expert ILRT Input Development	3:00–5:00 pm
Day 3 – Morning Session		
	Discussion of ILRT Failure Probability Results	8:30–9:00 am
	Discussion of ILRT Risk Impact Results	9:00–9:30 am
	Meeting Conclusion	9:30–10:00 am

C.2.1 Expert Elicitation Meeting: Day 1 – Morning Session

In the Day 1 morning session, the topics presented include an introduction, a presentation of the problem statement, presentation of the expert elicitation process, and expert panel training. Except for the training, the material included in these presentations is familiar to the experts because they will have been provided all preparation materials as part of the expert elicitation preparation.

The expert panel elicitation meeting begins with a 30-minute introduction. During this period, the experts are introduced to each other, and the goals and objectives of the expert elicitation are provided.

In the first presentation, the problem statement is reviewed. This material has already been provided as part of the expert elicitation preparation material. It is presented and reviewed with the experts.

In the second presentation, an overview of the expert panel elicitation process is provided. As in the case with the problem statement, experts are familiar with the material because it was provided as part of the preparation package. This presentation serves as a primer for the last presentation of the morning session, which is the expert elicitation training session.

During the two-hour expert elicitation training session, experts are provided training on the details of the expert elicitation process. The details include information on potential bias mechanisms and an in-class exercise of “almanac”-type questions designed to illustrate bias mechanisms.

C.2.2 Expert Elicitation Meeting: Day 1 – Afternoon Session

In the afternoon session, the topics presented include probabilistic safety assessment (PSA) concepts, application of PSA concepts to ILRT optimization, presentation of containment degradation events and mechanisms, and the ILRT database and other relevant data.

The first presentation of the afternoon session is a presentation on PSA concepts. This presentation is an overview of basic concepts of probabilistic safety assessment.

The second presentation of the afternoon session is on the application of the PSA concepts to the assessment of the risk impact associated with the optimization of ILRT intervals. Specifically, both methods employed to determine the risk impact and the role of expert elicitation are discussed.

The third presentation of the afternoon session covers containment degradation events and mechanisms. This presentation is a primer for the final presentation of the day.

The final presentation of Day 1 covers the ILRT events database and other relevant data. The process of the collection of the events, the availability of additional information, and the preliminary sorting of the data are also discussed.

C.2.3 Expert Elicitation Meeting: Day 2 – Morning Session

In the Day 2 morning session, the topics presented include a review of expert training and ILRT database, presentation of the expert elicitation example, and expert discussion of ILRT issues.

The morning session of Day 2 begins with a review of the expert elicitation training and the ILRT database.

The second presentation is the expert elicitation example. In this example, the use of the expert elicitation gathered information is demonstrated. This demonstration includes the assessment of the ILRT failure probability and the resulting effect of that failure probability on the assessment of the risk impact associated with the optimization of the ILRT Type A testing intervals.

The third presentation of the morning session is the discussion of ILRT issues. This discussion includes, but is not limited to, discussion of the potential containment failures’ modes and causes. The failure modes include those that have been experienced in the data as well as those potential failure modes that have not yet been experienced. Also included in the presentation will be an actual database of found degradations, some commonly found during in-service inspections (such as corrosion of liner plates or steel shell near moisture barriers), and some that are found after a number of years of hibernation (concealed corrosion).

C.2.4 Expert Elicitation Meeting: Day 2 – Afternoon Session

The afternoon session begins with the continuation of the discussion on the ILRT issues.

The second presentation of the afternoon session is the solicitation of the experts’ individual opinions. The expert solicitation is performed using the form contained in Appendix B to this report. This is the first part of the expert opinion elicitation. Following the collection of the expert opinion, the individual expert opinions are shared and discussed. The presentation ends with the submission of the final individual expert opinions. The individual expert opinions are combined to produce the common community distribution. The community distribution is developed by the technical integrator. The community distribution is presented to the experts on the morning of Day 3.

C.2.5 Expert Elicitation Meeting: Day 3 – Morning Session

On the morning of the third day, the community distribution is presented to the experts. The community distribution is discussed in detail, including the significant contributors to the distribution and the resulting risk impact associated with the ILRT testing interval optimization.

During the discussion of the community distribution and risk impact assessment results, feedback from the experts is solicited. Any changes to the community distribution and the resulting impacts on the ILRT testing interval optimization are presented to the experts.

Experts are finally asked for “buy-in” to their personal inputs, the resulting community distribution, and the resulting risk impact assessment from the optimization of ILRT testing intervals.

C.3 Steering Committee Review

Following completion of the expert elicitation, the NEI ILRT task force will be given the draft report, including the results of the expert elicitation and the results of the risk impact assessment of the ILRT testing optimization for review. This review is intended to provide a broad overview of the processes employed and industry-wide results of the risk impact assessment of ILRT interval extension optimization.

C.4 Expert Elicitation Input Form

The attached expert elicitation input table presents the form and type of input requested from the experts. The input from the experts is requested in tabular format. The table is described in detail in the following report sections.

In summary, the experts are asked to complete the table based on 1000 hypothetical tests. The experts are requested to augment the table with additional failure modes that may not appear on the table. Special attention to the effects of aging on potential containment failure modes is emphasized.

Fractions as well as whole numbers can be used in the table entries. For example, a fraction of 0.1 indicates that this failure mode would be expected once per 10,000 tests. A fraction of 0.01 indicates that this failure mode would be experienced once per 100,000 tests.

From the ILRT database, an initial attempt is made to complete the table. Because only small ILRT degradations have occurred, the entries on the table are limited. Experts are asked to augment the current small containment leakage columns. The initial attempt to complete the table is performed: it is preferable to elicit relative rather than absolute values from the experts because people are generally more comfortable making comparisons than estimating frequencies for phenomena with which they have little or no experience.

Therefore, for small leakage pathways, frequencies relative to failure mode frequencies for which data are available are elicited. For example, if few data are available for design deficiencies, ask the experts to estimate the ratio of the design deficiency frequency to the corrosion frequency.

The same process is applied to the elicitation of frequencies for medium-leakage pathways. That is, for medium-leakage pathways, frequencies relative to the corresponding frequencies for small leaks for the same failure mode are elicited. For large leaks, frequencies relative to medium-leak frequencies for the same failure mode are elicited.

C.4.1 Summary of Expert Elicitation Input Table Description

Table C-2 shows the summary of the expert elicitation input. Column 1 of the table, “No.,” is the numerical entry number.

Column 2 of the table, “Failure Mode or Degradation Description,” presents a potential failure mode of the ILRT. The majority of entries in this column are taken from the ILRT database representing previous linear degradations or leakage pathways. Other potential ILRT failure modes or containment degradation modes are also listed whether they have been experienced in the data or not. Blank lines are provided for experts to add additional containment degradation mechanisms not listed in the table. These additional failure modes or containment degradation events are discussed among the experts during the various expert elicitation discussion sessions.

Column 3, “Estimate of Low, Best, and High Value,” presents the characterization of the estimate provided by the experts. That is, for each containment failure classification (small, medium, large, and extremely large), the experts are requested to provide a “best” estimate as well as a low and high value relative to the “best” estimate.

Column 4, “Small Leakage Pathway,” is composed of three sub-columns (4a, 4b, and 4c). These sub-columns are described in detail below.

Column 4a, “Small Leakage Pathway – Total Degraded ILRTs,” presents the total number or fraction of events for each containment degradation or containment leakage pathway that the experts feel could result in a small leakage pathway. The number or fraction of degraded events should represent the number of events out of 1000 containment degradations discovered either through the ILRT, containment inspections, or other means. A small leakage pathway is defined as a leakage pathway that would result in an La of 1 or greater and less than 2 La. In addition, experts are asked to augment column 2 with any additional failure modes or containment degradations that do not currently appear in the table.

On the spreadsheet containing the historical ILRT data, the number of events from the ILRT database is a ratio that represents the number of failures in 1000 tests for each containment degradation or failure mode. It is conservatively assumed that the ILRT database was representative of approximately 400 successful tests. Therefore, the number of events was multiplied by 2.5 so that the result represented the number of events out of 1000 hypothetical tests.

Column 4b, “Small Leakage Pathways – Detected by Alternate Means,” presents the number or fraction of ILRT events for each containment degradation or containment leakage pathway that is small and that the experts feel could be detected or discovered by alternate means. Detection by alternate means includes other inspections, normal operation, or other tests, such as a local leak rate test. This column can include a fraction that is thought would be detected. The experts are asked to complete or change this column. As with the other columns in this table, it is to be based on 1000 ILRTs performed and entries can be in fractional form.

On the spreadsheet containing the historical ILRT data, the number of small-leakage events that were detected by alternate means is a ratio that represents the number of detections per 1000 ILRTs performed.

Column 4c, “Small Leakage Pathway – Detectable by ILRT Only (failures),” represents those leakage path events identified in the course of conducting ILRTs or that could only be detected by an ILRT Type A test. This value is calculated by subtracting the detected events from the total number of events (subtract column 4b from 4a). The resulting value is used in the estimation of the risk impact associated with the optimization of ILRT testing intervals, because these leakage path events represent those detectable only during the conduct of an ILRT.

Column 5, “Medium Leakage Pathway,” is composed of three sub-columns (5a, 5b, and 5c). These three sub-columns’ descriptions are similar to the above for the small leakage pathway, except that a medium pathway is defined as a leakage pathway that would result in an La from 2 to <10 La.

Column 6, “Large Leakage Pathway,” is composed of three sub-columns (6a, 6b, and 6c). The three sub-columns’ descriptions are similar to the above for the small leakage pathway, except that a large pathway is defined as a leakage pathway that would result in an La of greater than 10 La.

Column 7, “Extremely Large Pathway,” is composed of three sub-columns (7a, 7b, and 7c). The three sub-columns’ descriptions are similar to the above for the small leakage pathway, except that an extremely large pathway is defined as a leakage pathway that would result in an La greater than 100 La. Experts should note that certain failure modes may not be applicable given the size of this postulated leakage path. Experts should note these cases in the comments section of the form.

Column 8, “Notes,” provides a space for the experts to provide a basis for the assigned values. Due to space limitations on the table, experts are asked to number their notes and comments and provide them on a separate lined form.

C.4.2 Summary of Expert Elicitation Input Table Rows

The rows in the expert elicitation input table are sequentially numbered. Each numbered entry represents a containment failure mode that can result in a containment leakage event. Some failure modes have been experienced in the ILRT database, and these appear on the table. Other containment failure modes have not been experienced and are hypothetical. Experts are encouraged, based on their experience, to augment or change the table with the deletion or addition of failure modes. Special consideration is given to those failure modes that are age-related and may appear in the current ILRT testing data.

A summary row is provided in the table. In this summary row, the contributions to small, medium, large, and extremely large containment degradations or failure modes are summed. In addition, those failure modes detected by alternate means are summed for the leakage classes of small, medium, large, and extremely large. Lastly, the same is performed for the total “Detectable by ILRT Only” columns for each size category.

The above report sections present the planned elicitation of expert opinion. The experts were free to change the process and/or inputs as they saw fit to account for all the potential contributors to the ILRT failure probability. The details of the experts’ changes to the process and input are provided in “Expert Elicitation Results and Analysis,” Section 6.

Table C-2
Summary of Expert Elicitation

No.	Failure Mode or degradation Description	Estimate of Low, Best, and High Value	Small Leakage Pathway (< 2La)			Medium Leakage Pathway (2 - 10 La)			Large Leakage Pathway (> 10 La)			Extremely Large Pathway (> 100 La)			Notes
			Total Degraded ILRTs	Detected by Alternate Means	Detectable by ILRT Only (failures)	Total Degraded ILRTs	Detected by Alternate Means	Detectable by ILRT Only (failures)	Total Degraded ILRTs	Detected by Alternate Means	Detectable by ILRT Only (failures)	Total Degraded ILRTs	Detected by Alternate Means	Detectable by ILRT Only (failures)	
1	Original containment design deficiency	Low													
		"Best"													
		High													
2	Construction error or deficiency (e.g., construction debris in concrete)	Low													
		"Best"													
		High													
3	Human error associated with testing or maintenance (e.g., testing equipment left on penetration, not replacing caps on containment pressure instruments, improper alignment of valve components, use of improper components such as o-rings, washers in mechanical joints)	Low													
		"Best"													
		High													
4	Human error, design error or other deficiency associated with modifications (e.g., purge valves installed in wrong direction, spare pipes not capped, debris left in isolation valve, etc.)	Low													
		"Best"													
		High													
5	Erosion	Low													
		"Best"													
		High													
6	Corrosion (e.g., corrosion near water interface in bilges, corrosion of expansion bellows, corrosion of pipe caps, etc.)	Low													
		"Best"													
		High													
7	Fatigue failures (e.g., bellows fatigue failure)	Low													
		"Best"													
		High													
8	Others / Unknown	Low													
		"Best"													
		High													
9	TBD	Low													
		"Best"													
		High													
	TOTALS	Low													
		"Best"													
		High													

D
EXPERT ELICITATION RESULTS AND ANALYSIS

This appendix provides the results of the expert elicitation as well as the analysis of those results. Included are the changes made by the experts to the input form and processes.

D.1 Expert Elicitation Input Changes

As part of the expert elicitation process, the experts are free to change the expert elicitation process and inputs based on their collective experience and judgment. As a result of expert deliberation, several changes were made to the expert elicitation form. These changes included the following:

- Development of separate input forms for the collection of containment failure modes based on containment size. Separate forms were developed to address large containment types as well as small containments. For the purposes of discussion, small containments were those less than a million cubic feet in free volume. Those containments larger than approximately one million cubic feet in free volume were considered large containments. In general, the small containments were those associated with certain BWRs and ice condenser containment designs. It was agreed by the experts to collect expert opinion on both containment designs and to decide based on statistical analysis whether significant differences existed to warrant the development of separate ILRT "failure" probabilities.
- The collection of expert opinion was based on the existing testing scheme that is present in the data. This is conservatively considered to be an ILRT every three years. While it was recognized that the data were indeed collected over a period where the ILRT testing frequency ranged from an average of once every three years (three ILRTs per 10-year frequency) to once per 10 years, the experts felt that the majority of testing data were obtained from the three-in-10-years ILRT testing frequency.
- Adjustment of the column for large leakage pathway from representing leakage of >10 La to a leakage of 10–100 La.
- Significant changes to failure modes were made by the experts. Specifically, a smaller number of failure modes were addressed in the input form based on the expert opinion that the current set of containment failure modes overlapped and potentially double-counted the potential containment failure modes. The failure modes (1) original containment design deficiency and (5) erosion were eliminated. Events initially assigned to these categories were re-categorized into the final "Tabulation and Categorization of Historical ILRT Data," Appendix A.

The revised expert elicitation input forms are displayed in Appendix B.

D.2 Expert Elicitation Input

The input received from the experts is presented in detail in Appendix B. The experts deliberated on all the facets of containment bypass pathways. The significant areas for deliberation included:

- The potential containment failure modes to be considered
- The effect of the failure modes on containment leakage
- The potential to detect excessive leak paths (failures) with tests, maintenance, and inspections, other than integrated leak rate testing
- The effects of aging on the containments and the resulting failure modes
- The fact that not all potential containment failure modes may appear in the current data (failure mode hibernation)
- Different containment types having the potential for different failure modes with potentially different failure rates

Following significant deliberation, the experts provided their individual input on the adjusted expert elicitation forms. The input from the experts is solicited in the following form.

As stated previously, input is elicited for four ranges of leakage pathways. These four ranges are presented in columns in the expert elicitation form. The four leakage pathways size ranges are as follows:

- From 1 La to <2 La
- From 2 La to <10 La
- From 10 La to 100 La [10]
- Greater than 100 La

Within each leakage pathway range, input is elicited on the potential for any containment bypass pathway of the specific size, potential to detect the leakage pathway by alternate means (including other testing), maintenance, inspections, and finally, the total containment bypass pathway that can only be detected by the performance of the ILRT. This input is presented in columns in the expert elicitation form under each leakage pathway range.

[10] The initial expert elicitation form contained the ranges of “>10 La” and “>100 La.” During the expert elicitation, these entries were clarified to “10–100 La” and “>100 La.”

For each of these leakage pathway ranges, the input is solicited by containment failure mode. The containment failure modes are presented in rows of the input elicitation form. A total of five containment failure modes were identified by the experts. These five failure modes are:

1. Construction errors or deficiency. An example is construction debris in concrete.
2. Human error associated with testing or maintenance. For example, testing equipment left on penetration, not replacing caps on containment pressure instruments, improper alignment of valve components, and/or improper components such as O-rings or washers in mechanical joints.
3. Human error, design error, or other deficiency associated with modifications. For example, purge valves installed in wrong direction, spare pipes not capped, and debris left in isolation valve.
4. Corrosion. For example, corrosion near water interface in bilges, corrosion of expansion bellows, and corrosion of pipe caps.
5. Fatigue failures. An example is bellows fatigue failure.

For each containment failure mode, the experts provided a low, “best,” and high estimate for the number of failures based on 1000 hypothetical tests. In addition, a row was added to the table that provides the totals for the potential for a containment bypass pathway within the specified range. These totals included a total of the potential for the failure, a detection of the failure by alternate means, and the potential that the bypass pathway can be detected only by the performance of an ILRT.

The experts completed this input for both small and large containments. The detailed expert input is contained in Appendix B.

D.3 Statistical Analysis of the Expert Elicitation Input

Given the large amount of input collected from the experts, it is necessary to perform analysis of their collective input to develop the community distribution. Specifically, the risk impact assessment of the ILRT interval optimization requires the determination of the ILRT “failure” rate as a function of containment leakage pathway.

D.3.1 Statistical Analysis – Introduction

The purpose of this analysis is to determine a relationship between the containment leak size determined by an ILRT and its probability of occurrence. Let *A* be a random variable denoting the containment leak size measured in La. The desired relationship is the complementary cumulative distribution function (CCDF) *Q(a)* of *A*, which is defined as:

$$Q(a) \equiv \Pr\{A \geq a\} \quad \textbf{Eq. D-1}$$

In this analysis, it is assumed that A has a Weibull distribution, which has been chosen because of its ability to assume a wide variety of shapes (both increasing and decreasing hazard rates) and mathematical convenience. In reliability engineering, the Weibull distribution is often used to model the breaking strengths of materials. The CCDF of the Weibull distribution is:

$$Q(a) \equiv \exp(-\lambda a^{\beta}) \qquad a, \lambda, \beta > 0 \tag{Eq. D-2}$$

The parameter λ is termed the *scale parameter*; the parameter β is termed the *shape parameter*. Thus, the objective of the statistical analysis is to estimate the parameters λ and β using the information obtained through the expert elicitation process.

Least squares estimation has been used to determine the values of the parameters λ and β. Equation (2) may be linearized using a double logarithmic transformation:

$$\ln\left[\ln\left(\frac{1}{Q(a)}\right)\right] = \ln\lambda + \beta\ln a \tag{Eq. D-3}$$

Assume that estimates of $Q_j = Q(a_j)$ exist for various containment leak sizes a_j. Define:

$$\begin{aligned} y_j &= \ln\left[\ln\left(\frac{1}{Q_j}\right)\right] \\ x_j &= \ln a_j \\ b_1 &= \ln\lambda \\ b_2 &= \beta \end{aligned} \tag{Eq. D-4}$$

Then, the parameters λ and β may be determined through solution of the linear regression model:

$$y = b_1 + b_2 x + \varepsilon \tag{Eq. D-5}$$

The quantity ε denotes a random quantity to account for the measurement error in each y_j value. In ordinary least squares estimation, it is assumed that:

- The measurement errors are independent across the yj values (the measurement error for a given yj value is independent of the measurement errors for all other yj values).
- The measurement errors are described by a common normal uncertainty distribution having variance σ2.

As discussed in the following paragraphs, neither of these assumptions holds. Therefore, a generalized least squares method must be used.

D.3.2 Statistical Analysis – Input Information

In general, each expert has estimated the probability that the containment leak size falls into one of four ranges:

$$\begin{aligned} P_1 &= \Pr\{1 < A \le 2\} \\ P_2 &= \Pr\{2 < A \le 10\} \\ P_3 &= \Pr\{10 < A \le 100\} \\ P_4 &= \Pr\{A > 100\} \end{aligned} \qquad \textbf{Eq. D-6}$$

Recognizing the uncertainties involved, the actual information provided by each expert consists of order triplets (P_{iL}, P_{iB}, P_{iH}) denoting the low, best, and high estimate of the various P values. Thus, the P values are random variables whose distributions must be determined by using the ordered triplets provided by each expert. It is assumed that the P values are independent random variables having the following parameters:

$$\begin{aligned} \mu_{Pi} &= \text{mean} \\ \sigma_{Pi}^2 &= \text{variance} \end{aligned}$$

The variance of each P value is estimated using Chebyshev's Inequality, which applies to all probability distributions:

$$\Pr\{P_{iL} < P_i < P_{iH}\} = \Pr\{\mu_{Pi} - k\sigma_{Pi} < P_i < \mu_{Pi} + k\sigma_{Pi}\} \ge 1 - \frac{1}{k_i^2} \qquad \textbf{Eq. D-7}$$

Thus:

$$\begin{aligned} P_{iL} &= \mu_{Pi} - k\sigma_{Pi} \\ P_{iB} &= \mu_{Pi} \\ P_{iH} &= \mu_{Pi} + k\sigma_{Pi} \end{aligned} \qquad \textbf{Eq. D-8}$$

Which suggests:

$$\begin{aligned} \mu_{Pi} &= P_{iB} \\ \sigma_{Pi}^2 &= \max\left[\frac{(P_{iB} - P_{iL})^2}{k^2}, \frac{(P_{iH} - P_{iB})^2}{k^2}\right] \end{aligned} \qquad \textbf{Eq. D-9}$$

The parameter k is related to the probability that P_i lies within the open interval (P_{iL}, P_{iH}). For example:

$$0.9 = \Pr\{P_{iL} < P_i < P_{iH}\} \geq 1 - \frac{1}{k^2} \quad \Rightarrow \quad k = \sqrt{\frac{1}{1-0.9}} = \sqrt{10} \qquad \textbf{Eq. D-10}$$

The P values relate to $Q(a)$ through the following equations:

$$\begin{aligned}
P_1 &= \Pr\{1 < A \leq 2\} = \Pr\{A > 1\} - \Pr\{A > 2\} = Q(1) - Q(2) \\
P_2 &= \Pr\{2 < A \leq 10\} = \Pr\{A > 2\} - \Pr(A > 10\} = Q(2) - Q(10) \\
P_3 &= \Pr\{10 < A \leq 100\} = \Pr\{A > 10\} - \Pr(A > 100\} = Q(10) - Q(100) \\
P_4 &= \Pr\{A > 100\} = Q(100)
\end{aligned} \qquad \textbf{Eq. D-11}$$

Rearranging the above equations shows that:

$$\begin{aligned}
Q_1 &= Q(1) = \Pr\{A > 1\} = P_1 + P_2 + P_3 + P_4 \\
Q_2 &= Q(2) = \Pr\{A > 2\} = P_2 + P_3 + P_4 \\
Q_3 &= Q(10) = \Pr\{A > 10\} = P_3 + P_4 \\
Q_4 &= Q(100) = \Pr\{A > 100\} = P_4
\end{aligned} \qquad \textbf{Eq. D-12}$$

Note that the Q values are dependent random variables because they are functions of the P values. In general, the Q values have different variances. Noting that the Q values are sums of independent random variables, then:

$$\begin{aligned}
\sigma_{Q1}^2 &= \sigma_{P1}^2 + \sigma_{P2}^2 + \sigma_{P3}^2 + \sigma_{P4}^2 \\
\sigma_{Q2}^2 &= \sigma_{P2}^2 + \sigma_{P3}^2 + \sigma_{P4}^2 \\
\sigma_{Q3}^2 &= \sigma_{P3}^2 + \sigma_{P4}^2 \\
\sigma_{Q4}^2 &= \sigma_{P4}^2 \\
\therefore \quad & \sigma_{Q1}^2 \neq \sigma_{Q2}^2 \neq \sigma_{Q3}^2 \neq \sigma_{Q4}^2
\end{aligned} \qquad \textbf{Eq. D-13}$$

The covariance between any two Q values is given by:

$$Cov(Q_i, Q_j) = \sum_{k=\max(i,j)}^{4} \sigma_k^2 > 0 \qquad \textbf{Eq. D-14}$$

D.3.3 Statistical Analysis – Generalized Least Squares Method

The generalized least squares method determines parameter estimates by minimizing the following quantity:

$$D^2 = \mathbf{e}'\mathbf{\Sigma}^{-1}\mathbf{e} \qquad \textbf{Eq. D-15}$$

Where D^2 is a weighted sum of the squared residuals. The "D" means deviation, and the "2" implies squared. The $\mathbf{e}$ is an n x 1 matrix (column vector) of the residuals ($e_j = y_j - b_1 - b_2\, x_j$), and $\mathbf{\Sigma}$ is an n x n covariance matrix that describes the measurement errors in the y_j values. For the superscripts, the prime denotes matrix transpose and the exponent -1 denotes matrix inversion. Define:

$$\mathbf{y} = \begin{bmatrix} y_1 \\ \\ y_n \end{bmatrix} \qquad \mathbf{x} = \begin{bmatrix} 1 & x_1 \\ & \\ 1 & x_n \end{bmatrix} \qquad \mathbf{b} = \begin{bmatrix} b_1 \\ b_2 \end{bmatrix} \qquad \textbf{Eq. D-16}$$

Then, the generalized least squares solution is given by:

$$\mathbf{b} = \left(\mathbf{x}'\mathbf{\Sigma}^{-1}\mathbf{x}\right)^{-1} \mathbf{x}'\mathbf{\Sigma}^{-1}\mathbf{y} \qquad \textbf{Eq. D-17}$$

The covariance matrix of the parameter estimates is given by:

$$Var(\mathbf{b}) = \left(\mathbf{x}'\mathbf{\Sigma}^{-1}\mathbf{x}\right)^{-1} = \begin{bmatrix} \sigma_{b1}^2 & \sigma_{b1b2}^2 \\ & \\ \sigma_{b1b2}^2 & \sigma_{b2}^2 \end{bmatrix} \qquad \textbf{Eq. D-18}$$

The $\mathbf{\Sigma}$ matrix is determined by considering the impact of the uncertainties of the P values on the y values. These impacts can be approximated using statistical error propagation (the "delta method"):

$$\sigma_{Yi}^2 \approx \sum_{k=1}^{4} \left(\frac{\partial Y_i}{\partial P_k}\right)^2 \sigma_{Pk}^2 \qquad \textit{variance terms}$$

$$\sigma_{YiYj}^2 \approx \sum_{k=1}^{4} \left(\frac{\partial Y_i}{\partial P_k}\right)\left(\frac{\partial Y_j}{\partial P_k}\right)\sigma_{Pk}^2 \qquad \textit{covariance terms} \qquad \textbf{Eq. D-19}$$

Where the partial derivatives are evaluated at the means of the P values. It is convenient to define:

$$\phi_k = \mu_{Qk} \ln \mu_{Qk} \qquad \textbf{Eq. D-20}$$

Then:

$$\mathbf{\Sigma} = \begin{bmatrix} \frac{\sigma_{Q1}^2}{\phi_1^2} & \frac{\sigma_{Q2}^2}{\phi_1\phi_2} & \frac{\sigma_{Q3}^2}{\phi_1\phi_3} & \frac{\sigma_{Q4}^2}{\phi_1\phi_4} \\ \frac{\sigma_{Q2}^2}{\phi_1\phi_2} & \frac{\sigma_{Q2}^2}{\phi_2^2} & \frac{\sigma_{Q3}^2}{\phi_2\phi_3} & \frac{\sigma_{Q4}^2}{\phi_2\phi_4} \\ \frac{\sigma_{Q3}^2}{\phi_1\phi_3} & \frac{\sigma_{Q3}^2}{\phi_2\phi_3} & \frac{\sigma_{Q3}^2}{\phi_3^2} & \frac{\sigma_{Q4}^2}{\phi_3\phi_4} \\ \frac{\sigma_{Q4}^2}{\phi_1\phi_4} & \frac{\sigma_{Q4}^2}{\phi_2\phi_4} & \frac{\sigma_{Q4}^2}{\phi_3\phi_4} & \frac{\sigma_{Q4}^2}{\phi_4^2} \end{bmatrix} \quad \textbf{Eq. D-21}$$

D.3.4 Statistical Analysis – Uncertainty Bounds

The generalized least squares parameter estimates and their associated covariance matrix are used to estimate *Q(a)* and its uncertainty bounds. The point estimate of *Q(a)* is given by:

$$\hat{Q}(a) = \exp\left\lfloor -\exp(\hat{b}_1 + \hat{b}_2 \ln a)\right\rfloor \quad \textbf{Eq. D-22}$$

Let X be a random variable defined as the logit transformation of *Q(a)*:

$$X = \operatorname{logit}(Q) = \ln\left(\frac{Q}{1-Q}\right) \quad \textbf{Eq. D-23}$$

It is assumed that X has a normal distribution, with mean μ_V and standard deviation σ_V. Using statistical error propagation, the parameters of V are given by:

$$\mu_V = \ln\left(\frac{\hat{Q}(a)}{1-\hat{Q}(a)}\right) \quad \textbf{Eq. D-24}$$

$$\sigma_V = \frac{\exp(\hat{b}_1 + \hat{b}_2 \ln a)}{1-\hat{Q}(a)}\sqrt{\hat{\sigma}_{b1}^2 + (\ln a)^2\hat{\sigma}_{b2}^2 + 2(\ln a)\hat{\sigma}_{b1b2}^2} \quad \textbf{Eq. D-25}$$

Applying Equations (23) through (25), it can be demonstrated that:

$$\hat{Q}_{0.05}(a) = \frac{\hat{Q}(a)}{\hat{Q}(a) + \left(1-\hat{Q}(a)\right)w}$$

$$\hat{Q}_{0.95}(a) = \frac{\hat{Q}(a)}{\hat{Q}(a) + \left(1-\hat{Q}(a)\right)\frac{1}{w}} \quad \textbf{Eq. D-26}$$

Where:

$$w = \exp(z_{0.95}\sigma_V) \qquad \textbf{Eq. D-27}$$

and $z_{0.95}$ is the 95th percentile of the standard normal distribution (≈1.645).

D.3.5 Statistical Analysis – Combining Expert Opinion

For a given leak size a, *Q(a)* has an associated uncertainty distribution. Define:

$$F(q) = \Pr\{Q(a) \le q\} \qquad \textbf{Eq. D-28}$$

That is, *F(q)* is the cumulative probability distribution function of *Q(a)*. Expert opinions have been aggregated by forming a mixture distribution of the *Q(a)* probability distributions developed for each expert:

$$F(q) = \frac{1}{n}\sum_{i=1}^{n} F_i(q) \qquad \textbf{Eq. D-29}$$

Where *F(q)* denotes the aggregated cumulative distribution of *Q(a)*, $F_I(q)$ denotes the cumulative distribution function of *Q(a)* developed from the information provided by the *i*th expert, and *n* denotes the number of experts. Explicitly:

$$F(q) = \frac{1}{n}\sum_{i=1}^{n} \Phi\left[\frac{logit(q) - (b_{1i} + b_{2i}\ln a)}{\sigma_{yi}}\right] \qquad \textbf{Eq. D-30}$$

Where Φ() denotes the standard normal cumulative distribution function. In order to determine percentiles of the aggregated distribution, Equation (30) must be solved numerically for *q* given that *F(q)* equals a specified value (for example, 0.05 or 0.95).

D.3.6 Statistical Analysis – Final Results

The detailed final results of the statistical analysis of the expert elicitation are provided in Appendix F. In summary, a spreadsheet and visual basic computer routines were developed to assist in the analysis of the input data. Table D-1 presents the results of the analysis of the expert elicited input.

Table D-1
Expert Elicitation Results – Leak Size Versus Probability

Leakage Size (La)	Mean Probability of Occurrence
1	2.65E-02
2	1.59E-02
5	7.42E-03
10	3.88E-03
20	1.88E-03
35	9.86E-04
50	6.33E-04
100	2.47E-04
200	8.57E-05
500	1.75E-05
600	1.24E-05
1000	4.50E-06
2000	1.01E-06
5000	1.11E-07
10000	1.73E-08

The input data used was the trim mean. That is, the lowest and highest experts were not included in the development of the community distribution. This treatment was performed for several reasons. One expert used zero several times in the assignment of the probability of ILRT failure. Zeros are difficult to treat in the statistical evaluation of the expert input. Therefore, this expert was not included in the development of the community distribution. Because the lowest expert was not included in the development of the community distribution, it was prudent to not include the highest expert in the development of the community distribution as well. This treatment results in the use of a set of four experts as opposed to six to develop the community distribution. Therefore, the community distribution represents the center of the input data collected.

In addition, no community distribution was developed for the small containment case. This is a result of the fact that analysis of the small containment input data actually produces slightly lower values for the probability of a leakage pathway in the small containments. The differences are very small and do not represent a significant difference in the probability. Therefore, the small containment case was not evaluated. It should be noted that one expert did not complete small containment input sheets because he or she believed that there was no reason to treat the small containments differently than the large containment type.

Both of the above treatments of the input data were discussed with experts during the elicitation meeting as being potential treatments of the final results. Experts agreed with this treatment. The final results of the determination of the probability of a leakage pathway can be described in tabular format, as follows.

Appendix C contains the detailed results of the expert elicitation. It is interesting to note that the values contained in Table D-1 agree relatively closely with those produced using other methods, such as those in the joint applications report for containment integrated leak rate test interval extension [15].

Table D-2 provides a comparison of the pre-existing leakage probabilities developed using various statistical techniques. The current Jeffreys Non-Informative Prior is based on 182 tests. These tests were limited to those utilities and nuclear units that responded to NEI surveys. It is estimated that approximately 400 ILRTs have been performed in the nuclear industry. For comparison purposes only, these values are presented on Table D-2.

Table D-2
Comparison of Pre-Existing Leakage Probabilities

Statistical Method	Statistical Method Value	Expert Elicited Value at 35 La	Percent Difference
Based on 182 Tests			
Chebychev	5.50E-03	9.86E-04	82%
Jeffreys Non-Informed Prior	2.70E-03	9.86E-04	63%
Typical Ranges	1.60E-03	9.86E-04	38%
	5.00E-04	9.86E-04	-97%
Based on 400 Tests			
Chebychev	2.50E-03	9.86E-04	61%
Jeffreys Non-Informed Prior	1.25E-03	9.86E-04	21%
Typical Ranges	7.50E-04	9.86E-04	-31%
	2.50E-04	9.86E-04	-294%

Table D-3 provides a rough order of magnitude estimation of the actual number of ILRTs performed for U.S. nuclear plants from 1977 through 2001. The estimate is based on the number of unit operating years for the interval in which three ILRTs were performed per 10 years for the years 1977–1994. From 1994 through 2001, the number of ILRTs performed is based on one ILRT per 10 unit operating year. No estimation is made for the number of ILRTs performed from 2001 to present.

Table D-3
Estimation of Actual Number of ILRTs Performed for Operating U.S. Plants

Year	Number of Units in Commercial Operation	Number of ILRTs (3/10 yrs - 1/3 test per unit-yr)	New Plants on Line	Adjusted ILRTs
77	57	17	5	12
78	64	19	7	12
79	67	20	3	17
80	68	20	1	19
81	70	21	2	19
82	74	22	4	18
83	75	22	1	21
84	78	23	3	20
85	82	24	4	20
86	90	27	8	19
87	96	28	6	22
88	102	30	6	24
89	107	32	5	27
90	110	33	3	30
91	111	33	1	32
92	110	33		33
93	106	31		31
94	107	32	1	34
Subtotal (through 1994):				410
Number of Tests 1995–2001:				38
TOTAL:				448

E
EXPERT ELICITATION INPUT DATA

This appendix presents a summary of the expert elicitation input. A total of eight tables are presented.

The first four tables are associated with the large containment type. A *large containment* was defined for the expert elicitation panel as a containment of greater than 1 million cubic feet of free volume. The four large containment type tables that are presented are the small leakage pathway (1–2 La), medium leakage pathway (2–10 La), large leakage pathway (10–100 La), and the extremely large leakage pathway (>100 La).

The second four tables are associated with small containments. A *small containment* was defined for the expert elicitation panel as a containment with less than 1 million cubic feet of volume. The four tables associated with the small containment type are the small leakage pathway (1–2 La), medium leakage pathway (2–10 La), large leakage pathway (10–100 La), and the extremely large leakage pathway (>100 La).

Each of the eight tables contains rows associated with the five containment failure modes identified by the expert elicitation panel as well as a total row. There are three major columns in each table. These major columns are the “Total Degraded ILRTs,” “Detected by Alternate Means,” and “Detectable by ILRT Only (Failures).” Each of the major columns has six minor columns. Each minor column represents a different expert’s input. The input is provided in the form of expected occurrences given 1000 hypothetical ILRTs.

Table E-1
Expert Elicitation Input – Large Containment with Small Leakage Pathway

No.	Failure Mode or Degradation Description	Estimate of Low, Best, and High Value	Small Leakage Pathway (1–2 La)																	
			Total Degraded ILRTs						Detected by Alternate Means						Detectable by ILRT Only (Failures)					
			a	b	c	d	e	f	a	b	c	d	e	f	a	b	c	d	e	f
1	Construction error or deficiency (for example, construction debris in concrete)	Low	1	5	0	1	0.1	5	0.9	4	0	1	0.1	4	0.1	0	0	0.1	0.1	0.01
		"Best"	10	10	0	10	7.5	8	9	7.5	0	7.5	5	6	1	2.5	0	2.5	2.5	2
		High	30	15	0	25	19	16	25	12	0	25	16	19	5	5	0	15	12	12
2	Human error associated with testing or maintenance (for example, testing equipment left on penetration, not replacing caps on containment pressure instruments, improper alignment of valve components, use of improper components such as O-rings, washers in mechanical joints)	Low	3	10	0	1	1	1	1.5	5	0	0.5	1	1	1.5	2	0	0.5	1	0.05
		"Best"	10	15	0	7.5	15	8	5	10	0	3.75	5	4	5	5	0	3.75	10	4
		High	30	22	0	25	31	25	15	15	0	20	16	16	15	7	0	20	23	16
3	Human error, design error or other deficiency associated with modifications (for example, purge valves installed in wrong direction, spare pipes not capped, debris left in isolation valve, and so on)	Low	5	5	0	2	1	1	2.5	4	0	1	0.1	1	2.5	2	0	2	1	0.05
		"Best"	12.5	10	0	12.5	12.5	10	6.25	7	0	2.5	2.5	4	6.25	3	0	10	10	6
		High	20	15	0	30	27	30	10	10	0	15	12	20	10	5	0	25	23	25
4	Corrosion (for example, corrosion near water interface in bilges, corrosion of expansion bellows, corrosion of pipe caps, and so on)	Low	2	5	0	5	1	1	1.8	5	0	2	1	1	0.2	2	0	1	0.1	0.05
		"Best"	10	15	0	15	10	10	9	10	0	10	7.5	8	1	5	0	5	2.5	2
		High	50	25	0	40	23	30	45	15	0	30	19	30	5	10	0	20	10	20
5	Fatigue failures (for example, bellows fatigue failure)	Low	0.5	1	0	1	0.1	0.01	0.4	0.5	0	1	0.1	0.01	0.1	0.2	0	0.1	0.01	0.01
		"Best"	2.5	3	0	2.5	2	0.1	2	2.5	0	2.5	1.9	0.05	0.5	0.5	0	0.5	0.1	0.05
		High	25	5	0	12	12	10	20	3	0	12	12	10	5	1	0	10	1	10
	TOTALS	Low	11.5	26	0	10	3.2	8.01	7.1	18.5	0	5.5	2.3	7.01	4.4	6.2	0	3.7	2.21	0.17
		"Best"	45	53	0	47.5	47	36.1	31.3	37	0	26.3	21.9	22.1	13.8	16	0	21.8	25.1	14.1
		High	155	82	0	132	112	111	115	55	0	102	75	95	40	28	0	90	69	83

Table E-2
Expert Elicitation Input – Large Containment with Medium Leakage Pathway

No.	Failure Mode or Degradation Description	Estimate of Low, Best, and High Value	Medium Leakage Pathway (2–10 La)																	
			Total Degraded ILRTs						Detected by Alternate Means						Detectable by ILRT Only (Failures)					
			a	b	c	d	e	f	a	B	c	d	e	f	a	b	c	d	e	f
1	Construction error or deficiency (for example, construction debris in concrete)	Low	0.2	2	0	0.1	0.1	2	0.18	0	0	0.1	0.1	1	0.02	0	0	0.1	0.1	0.1
		"Best"	2	5	0	5	2.5	3	1.8	2.5	0	2.5	1.5	2	0.2	2.5	0	2.5	1	1
		High	20	10	0	20	12	16	18	5	0	12	10	12	2	5	0	12	9	10
2	Human error associated with testing or maintenance (for example, testing equipment left on penetration, not replacing caps on containment pressure instruments, improper alignment of valve components, use of improper components such as O-rings, washers in mechanical joints)	Low	0.2	0	0	0.1	1	1	0.1	0	0	0.1	0.1	0.05	0.1	0	0	0.1	0.1	0.2
		"Best"	1	0	0	1.25	3	4	0.5	0	0	1.25	1.5	2	0.5	0	0	1	1.5	2
		High	5	2	0	15	14	20	2.5	2	0	15	10	20	2.5	0	0	10	10	20
3	Human error, design error or other deficiency associated with modifications (for example, purge valves installed in wrong direction, spare pipes not capped, debris left in isolation valve, and so on)	Low	0.2	0	0	0.1	1	0.1	0.1	0	0	0.1	0.1	0.1	0.1	0	0	0.1	0.1	0.1
		"Best"	1	0	0	1.25	3	2	0.5	0	0	1	1.5	1	0.5	0	0	1.25	1.5	1
		High	5	3	0	10	14	15	2.5	2	0	5	10	10	2.5	1	0	10	10	10
4	Corrosion (for example, corrosion near water interface in bilges, corrosion of expansion bellows, corrosion of pipe caps, and so on)	Low	0.1	0	0	0.1	0.1	0.5	0.09	0	0	0.1	0.1	0.5	0.01	0	0	0.1	0.1	0.25
		"Best"	1	1	0	5	2.5	5	0.9	1	0	2.5	1.5	4	0.1	0	0	2.5	1	1
		High	10	3	0	20	10	20	9	2	0	12	5	20	1	1	0	12	5	20
5	Fatigue failures (for example, bellows fatigue failure)	Low	0.01	0	0	0.01	0.01	0.005	9E-3	0	0	0.01	0.01	0.001	1E-3	0	0	0.01	0.001	0.001
		"Best"	0.1	0	0	0.1	0.1	0.01	9E-2	0	0	0.05	0.09	0.005	1E-2	0	0	0.05	0.01	0.005
		High	1	0	0	2	1	8	0.9	0	0	1	1	8	1E-1	0	0	1	0.1	8
	TOTALS	Low	0.71	2	0	0.41	2.21	3.61	0.48	0	0	0.41	0.41	1.65	0.23	0	0	0.41	0.4	0.65
		"Best"	5.1	6	0	12.6	11.1	14	3.79	3.5	0	7.3	6.09	9.01	1.31	2.5	0	7.3	5.01	5.01
		High	41	18	0	67	51	79	32.9	11	0	45	36	70	8.1	7	0	45	34.1	68

Table E-3
Expert Elicitation Input – Large Containment with Large Leakage Pathway

No.	Failure Mode or Degradation Description	Estimate of Low, Best, and High Value	Large Leakage Pathway (>10 La)																	
			Total Degraded ILRTs						Detected by Alternate Means						Detectable by ILRT Only (Failures)					
			a	b	c	d	e	f	a	b	c	d	e	f	a	b	c	d	e	f
1	Construction error or deficiency (for example, construction debris in concrete)	Low	0.01	0	0	0.1	0.1	0.05	0.009	0	0	0.1	0.1	0.25	1E-3	0	0	0.1	0.1	0.1
		"Best"	0.1	0	0	2.5	1.5	2	0.09	0	0	1.25	1	1	1E-2	0	0	1.25	0.5	1
		High	1	5	0	12	10	15	0.1	2.5	0	8	7	15	0.1	2.5	0	8	6	20
2	Human error associated with testing or maintenance (for example, testing equipment left on penetration, not replacing caps on containment pressure instruments, improper alignment of valve components, use of improper components such as O-rings, washers in mechanical joints)	Low	0.01	0	0	0.1	0.1	0.05	0.005	0	0	0.1	0.1	0.25	0.005	0	0	0.1	0.1	0.1
		"Best"	0.1	0	0	1.25	1.5	1	0.05	0	0	1	1	0.5	0.05	0	0	1.25	0.5	0.5
		High	1	0	0	15	10	16	0.5	0	0	10	10	12	0.5	0	0	15	10	12
3	Human error, design error or other deficiency associated with modifications (for example, purge valves installed in wrong direction, spare pipes not capped, debris left in isolation valve, and so on)	Low	0.01	0	0	0.1	0.1	0.01	0.01	0	0	0.01	0.1	0.01	0.01	0	0	0.01	0.1	0.1
		"Best"	0.05	0	0	1	1.5	1	0.05	0	0	0.5	1	0.05	0.05	0	0	0.5	0.5	0.5
		High	0.5	0	0	10	10	15	0.5	0	0	5	10	12	0.5	0	0	5	10	12
4	Corrosion (for example, corrosion near water interface in bilges, corrosion of expansion bellows, corrosion of pipe caps, and so on)	Low	0.01	0	0	0.1	0.1	0.5	9E-3	0	0	0.1	0.1	0.25	1E-3	0	0	0.1	0.1	0.1
		"Best"	0.1	0	0	2.5	1	2	9E-2	0	0	1.25	0.75	1	1E-2	0	0	1.25	0.25	1
		High	1	0	0	12	5	15	0.9	0	0	8	5	15	1E-2	0	0	8	3	20
5	Fatigue failures (for example, bellows fatigue failure)	Low	1E-3	0	0	0.01	0.001	1E-4	9E-4	0	0	0.01	0.001	1E-4	1E-4	0	0	0.01	1E-4	1E-4
		"Best"	1E-2	0	0	0.1	0.01	1E-3	9E-3	0	0	0.05	0.01	1E-3	1E-3	0	0	0.05	0.001	5E-4
		High	0.1	0	0	2	0.1	8	9E-2	0	0	1	0.1	8	1E-2	0	0	1	0.01	8
	TOTALS	Low	0.04	0	0	0.41	0.4	0.61	0.03	0	0	0.32	0.4	0.76	0.01	0	0	0.32	0.4	0.4
		"Best"	0.36	0	0	7.35	5.51	6	0.29	0	0	4.05	3.76	2.55	0.12	0	0	4.3	1.75	3
		High	3.6	5	0	51	35.1	69	2.09	2.5	0	32	32.1	62	1.12	2.5	0	37	29	72

Table E-4
Expert Elicitation Input – Large Containment with Extremely Large Leakage Pathway

No.	Failure Mode or Degradation Description	Estimate of Low, Best, and High Value	Extremely Large Pathway (>100 La)																	
			Total Degraded ILRTs						Detected by Alternate Means						Detectable by ILRT Only (Failures)					
			a	b	c	d	e	f	a	b	c	d	e	f	a	B	c	d	e	f
1	Construction error or deficiency (for example, construction debris in concrete)	Low	1E-4	0	0	0.01	0.1	0.1	9E-5	0	0	0.01	0.1	0.05	1E-5	0	0	0.01	0.1	0.05
		"Best"	1E-3	0	0	0.1	0.5	0.5	9E-4	0	0	0.05	0.25	0.25	1E-4	0	0	0.05	0.25	0.25
		High	1E-2	0	0	5	5	12	9E-3	0	0	2	4	15	1E-3	0	0	2	3.00	15
2	Human error associated with testing or maintenance (for example, testing equipment left on penetration, not replacing caps on containment pressure instruments, improper alignment of valve components, use of improper components such as O-rings, washers in mechanical joints)	Low	1E-4	0	0	0.001	0.1	0.05	9E-5	0	0	0.001	0.1	0.05	1E-5	0	0	0.001	0.10	0.001
		"Best"	1E-3	0	0	0.01	0.5	0.25	9E-4	0	0	0.01	0.25	0.2	1E-4	0	0	0.01	0.25	0.1
		High	1E-2	0	0	1	10	12	9E-3	0	0	1	10	12	1E-3	0	0	1	10.00	15
3	Human error, design error or other deficiency associated with modifications (for example, purge valves installed in wrong direction, spare pipes not capped, debris left in isolation valve, and so on)	Low	1E-4	0	0	0.001	0.1	0.05	9E-5	0	0	0.001	0.1	0.05	1E-5	0	0	0.001	0.10	0.05
		"Best"	1E-3	0	0	0.01	0.5	0.25	9E-4	0	0	0.01	0.25	0.2	1E-4	0	0	0.01	0.25	0.1
		High	1E-2	0	0	1	10	15	9E-3	0	0	1	10	15	1E-3	0	0	1	10.00	15
4	Corrosion (for example, corrosion near water interface in bilges, corrosion of expansion bellows, corrosion of pipe caps, and so on)	Low	1E-4	0	0	0.01	0.01	0.1	9E-5	0	0	0.01	0.01	0.05	1E-5	0	0	0.01	0.001	0.05
		"Best"	1E-3	0	0	0.1	0.1	0.5	9E-4	0	0	0.05	0.075	0.25	1E-4	0	0	0.05	0.03	0.25
		High	1E-2	0	0	5	1	12	9E-3	0	0	2	1	15	1E-3	0	0	2	0.10	15
5	Fatigue failures (for example, bellows fatigue failure)	Low	1E-4	0	0	0.001	0.001	1E-4	9E-5	0	0	0.001	0.001	1E-5	1E-5	0	0	0.001	1E-4	1E-4
		"Best"	1E-3	0	0	0.01	0.01	1E-4	9E-4	0	0	0.01	0.01	5E-5	1E-4	0	0	0.01	0.001	5E-4
		High	1E-2	0	0	5	0.1	8	1E-3	0	0	0.5	0.1	8	1E-4	0	0	0.5	0.01	8
	TOTALS	Low	5E-4	0	0	0.02	0.31	0.3	5E-4	0	0	0.02	0.31	0.2	5E-5	0	0	0.02	0.3	0.15
		"Best"	0.01	0	0	0.23	1.61	1.5	5E-3	0	0	0.13	0.83	0.9	5E-4	0	0	0.13	0.78	0.7
		High	0.05	0	0	17	26.1	59	0.04	0	0	6.5	25.1	65	4E-3	0	0	6.5	23.1	68

Table E-5
Expert Elicitation Input – Small Containment with Small Leakage Pathway

No.	Failure Mode or Degradation Description	Estimate of Low, Best, and High Value	Small Leakage Pathway (1–2 La)																	
			Total Degraded ILRTs						Detected by Alternate Means						Detectable by ILRT Only (Failures)					
			a	b	c	d	e	f	a	b	c	d	e	f	A	b	c	d	e	f
1	Construction error or deficiency (for example, construction debris in concrete)	Low	1	5	25	1	0.1	5	0.9	3	2	1	0.1	4	0.1	0	0.5	0.1	0.1	0.1
		"Best"	10	10	15	10	2.5	8	9	5	13	7.5	1.5	6	1	5	2	2.5	1	2
		High	30	15	65	25	12	16	25	7	50	25	10	19	5	10	15	15	9	12
2	Human error associated with testing or maintenance (for example, testing equipment left on penetration, not replacing caps on containment pressure instruments, improper alignment of valve components, use of improper components such as O-rings, washers in mechanical joints)	Low	3	5	2	1	1	1	0.75	3	1.5	0.5	0.1	1	0.75	1	0.5	0.5	1	0.5
		"Best"	10	10	7	7.5	7.5	4	5	7.5	7	3.75	2.5	2	5	2.5	3	3.75	5	2
		High	30	15	25	25	35	20	15	10	15	20	25	20	15	5	10	20	35	25
3	Human error, design error or other deficiency associated with modifications (for example, purge valves installed in wrong direction, spare pipes not capped, debris left in isolation valve, and so on)	Low	5	5	0.8	2	1	1	2.5	4	0.2	1	0.1	1	2.5	2	0.6	2	1	0.5
		"Best"	12.5	10	15	12.5	12.5	10	6.25	7	7	2.5	2.5	4	6.25	3	8	10	10	6
		High	20	15	30	30	45	30	10	10	12	15	25	20	10	5	18	25	40	25
4	Corrosion (for example, corrosion near water interface in bilges, corrosion of expansion bellows, corrosion of pipe caps, and so on)	Low	1	10	2	5	1	1	0.9	10	1.5	2	1	1	0.1	2	0.5	1	0.1	0.5
		"Best"	5	20	25	15	5	5	4.5	15	20	10	4	4	0.5	5	8	5	1	1
		High	25	30	60	40	16	20	22.5	20	50	30	14	20	2.5	10	10	20	9	20
5	Fatigue failures (for example, bellows fatigue failure)	Low	0.5	1	0.4	1	0.1	0.01	0.4	0.5	0.2	1	0.1	0.01	0.1	0.2	0.2	0.1	0.01	0.01
		"Best"	2.5	3	2	2.5	2	0.1	2	2.5	1.2	2.5	1.9	0.05	0.5	0.5	8	0.5	0.1	0.05
		High	25	5	8	12	12	10	20	3	5	12	12	10	5	1	3	10	1	10
	TOTALS	Low	10.5	26	30.2	10	3.2	8.01	5.45	20.5	5.4	5.5	1.4	7.01	3.55	5.2	2.3	3.7	2.21	1.61
		"Best"	40	53	64	47.5	29.5	27.1	26.8	37	48.2	26.3	12.4	16.1	13.3	16	29	21.8	17.1	11.1
		High	130	80	188	132	120	96	92.5	50	132	102	86	89	37.5	31	56	90	94	92

Table E-6
Expert Elicitation Input – Small Containment with Medium Leakage Pathway

No.	Failure Mode or Degradation Description	Estimate of Low, Best, and High Value	Medium Leakage Pathway (2–10 La)																	
			Total Degraded ILRTs						Detected by Alternate Means						Detectable by ILRT Only (Failures)					
			a	b	c	d	e	f	a	b	c	d	e	f	a	b	c	d	e	f
1	Construction error or deficiency (for example, construction debris in concrete)	Low	0.2	2	2	0.1	0.1	2	0.18	0	1.5	0.1	0.1	1	0.02	0	0.5	0.1	0.1	0.1
		"Best"	2	5	5	5	1	3	1.8	2.5	4	2.5	0.5	2	0.2	2.5	1	2.5	0.5	1
		High	20	10	20	20	9	16	18	5	16	12	6	12	2	5	4	12	6	10
2	Human error associated with testing or maintenance (for example, testing equipment left on penetration, not replacing caps on containment pressure instruments, improper alignment of valve components, use of improper components such as O-rings, washers in mechanical joints)	Low	0.5	1	1.5	0.1	1	0.5	0.25	1	1.3	0.1	0.1	0.5	0.25	0	0.2	0.1	0.1	0.05
		"Best"	2.5	2.5	7	1.25	5	1	1.25	2.5	4	1.25	2.5	1	1.25	0	3	1	2.5	0.1
		High	10	7.5	20	15	35	25	10	5	16	15	25	25	5	2.5	4	10	25	15
3	Human error, design error or other deficiency associated with modifications (for example, purge valves installed in wrong direction, spare pipes not capped, debris left in isolation valve, and so on)	Low	0.5	0	0.5	0.1	1	0.1	0.25	0	0.2	0.1	0.1	0.1	0.25	0.1	0.3	0.1	0.1	0.1
		"Best"	2.5	2.5	12	1.25	5	2	1.25	2	5	1	2.5	1	1.25	0.5	7	1.25	2.5	1
		High	10	5	20	10	35	15	5	4	8	5	25	10	5	1	12	10	25	10
4	Corrosion (for example, corrosion near water interface in bilges, corrosion of expansion bellows, corrosion of pipe caps, and so on)	Low	0.1	0	1.8	0.1	0.1	0.5	0.09	0	1.4	0.1	0.1	0.5	0.01	0	0.4	0.1	0.01	0.25
		"Best"	1	2	20	5	1	2	0.9	2	17	2.5	0.75	1	0.1	0	3	2.5	0.25	1
		High	10	3	45	20	5	15	9	2	30	12	5	15	1	1	15	12	3	20
5	Fatigue failures (for example, bellows fatigue failure)	Low	0.01	0	0.3	0.01	0.01	0.005	9E-3	0	0.2	0.01	0.01	0.001	1E-3	0	0.1	0.01	0	0.001
		"Best"	0.1	0	1.8	0.1	0.1	0.01	9E-2	0	1.1	0.05	0.09	0.005	1E-2	0	7	0.05	0.01	0.005
		High	1	0	7	2	1	8	9E-1	0	5	1	1	8	1E-1	0	2	1	0.1	8
	TOTALS	Low	1.31	3	6.1	0.41	2.21	3.11	0.78	1	4.6	0.41	0.41	2.1	0.53	0.1	1.5	0.41	0.31	0.5
		"Best"	8.1	12	45.8	12.6	12.1	8.01	5.29	9	31.1	7.3	6.34	5.01	2.81	3	21	7.3	5.76	3.11
		High	51	25.5	112	67	85	79	42.9	16	75	45	62	70	13.1	9.5	37	45	59.1	63

Table E-7
Expert Elicitation Input – Small Containment with Large Leakage Pathway

No.	Failure Mode or Degradation Description	Estimate of Low, Best, and High Value	Large Leakage Pathway (>10 La to 100La)																	
			Total Degraded ILRTs						Detected by Alternate Means						Detectable by ILRT Only (Failures)					
			a	b	c	d	e	f	a	b	c	d	e	f	a	b	c	d	e	f
1	Construction error or deficiency (for example, construction debris in concrete)	Low	0.01	0	1	0.1	0.1	0.5	0.01	0	0.8	0.1	0.1	0.25	1.E-3	0	0.2	0.1	0.1	0.1
		"Best"	0.1	0	3	2.5	0.5	2	0.09	0	2	1.25	0.25	1	1.E-2	0	1	1.25	0.25	1
		High	1	5	10	12	6	15	0.1	2.5	8	8	4	15	0.1	2.5	2	8	4	20
2	Human error associated with testing or maintenance (for example, testing equipment left on penetration, not replacing caps on containment pressure instruments, improper alignment of valve components, use of improper components such as O-rings, washers in mechanical joints)	Low	0.5	0	0.8	0.1	0.1	0.1	0.25	0	0.5	0.1	0.1	0.1	0.25	0	0.3	0.1	0.1	0.05
		"Best"	2.5	0	2	1.25	2.5	0.5	1.25	0	1.8	1	1	0.5	1.25	0	0.2	1.25	1.5	0.1
		High	10	1	10	15	25	12	5	1	6	10	10	12	5	0	4	15	15	12
3	Human error, design error or other deficiency associated with modifications (for example, purge valves installed in wrong direction, spare pipes not capped, debris left in isolation valve, and so on)	Low	0.1	0	0.3	0.1	0.1	0.1	0.05	0	0.1	0.01	0.1	0.1	0.05	0	0.2	0.01	0.1	0.1
		"Best"	1	0	4	1	2.5	1	0.5	0	1.5	0.5	1	0.5	0.5	0	2.5	0.5	1.5	0.5
		High	10	0	15	10	25	15	5	0	7	5	10	15	5	0	8	5	15	15
4	Corrosion (for example, corrosion near water interface in bilges, corrosion of expansion bellows, corrosion of pipe caps, and so on)	Low	0.01	0	1.1	0.1	0.1	0.1	9E-3	0	0.8	0.1	0.01	0.1	1E-3	0	0.3	0.1	0.01	0.05
		"Best"	0.1	0	6	2.5	0.5	0.5	9E-2	0	4	1.25	0.350	0.25	1E-2	0	2	1.25	0.15	0.25
		High	1	0	20	12	4	15	9E-1	0	15	8	4	15	1E-1	0	5	8	2	1.5
5	Fatigue failures (for example, bellows fatigue failure)	Low	1E-3	0	0.2	0.01	1E-3	1E-4	9E-4	0	0.1	0.01	0.001	1E-4	1E-4	0	0.1	0.01	1E-4	1E-4
		"Best"	1E-2	0	1.2	0.1	0.01	0.001	9E-3	0	0.8	0.05	0.01	0.001	1E-3	0	0.4	0.05	1E-3	5E-4
		High	1E-1	0	6	2	0.1	8	9E-2	0	4	1	0.1	8	1E-2	0	2	1	0.01	8
	TOTALS	Low	0.62	0	3.4	0.41	0.4	0.8	0.32	0	2.3	0.32	0.31	0.55	0.3	0	1.1	0.32	0.31	0.3
		"Best"	3.71	0	16.2	7.35	6.01	4	1.94	0	10.1	4.05	2.61	2.25	1.77	0	6.1	4.3	3.4	1.85
		High	22.1	6	61	51	60.1	65	11.1	3.5	40	32	28.1	65	10.2	2.5	21	37	36	56.5

Table E-8
Expert Elicitation Input – Small Containment with Extremely Large Leakage Pathway

No.	Failure Mode or Degradation Description	Estimate of Low, Best, and High Value	Extremely Large Leakage Pathway (>100 La)																	
			Total Degraded ILRTs						Detected by Alternate Means						Detectable by ILRT Only (Failures)					
			a	b	c	d	e	f	a	b	c	d	e	f	a	b	c	d	e	f
1	Construction error or deficiency (for example, construction debris in concrete)	Low	1.E-4	0	0.5	0.01	0.1	0.1	9.E-5	0	0.3	0.01	0.01	0.05	1.E-5	0	0.2	0.01	0.01	0.05
		"Best"	1.E-3	0	1	0.1	0.25	0.5	9.E-4	0	0.7	0.05	0.1	0.25	1.E-4	0	0.3	0.05	0.15	0.25
		High	1.E-2	0	5	5	4	12	9.E-3	0	3	2	3	15	1.E-3	0	2	2	2	15
2	Human error associated with testing or maintenance (for example, testing equipment left on penetration, not replacing caps on containment pressure instruments, improper alignment of valve components, use of improper components such as O-rings, washers in mechanical joints)	Low	1E-3	0	0.6	0	0.1	0.05	9E-4	0	0.2	0.001	0.1	0.05	1E-4	0	0.4	0.001	0.1	0.01
		"Best"	1E-2	0	1	0.01	1	0.1	9E-3	0	0.4	0.01	0.5	0.1	1E-3	0	0.6	0.01	0.5	0.05
		High	1E-1	0	8	1	10	12	9E-2	0	3	1	10	12	1E-2	0	8	1	10	12
3	Human error, design error or other deficiency associated with modifications (for example, purge valves installed in wrong direction, spare pipes not capped, debris left in isolation valve, and so on)	Low	1E-3	0	0.2	0	0.1	0.05	9E-4	0	0.1	0.001	0.1	0.05	1E-4	0	0.1	0.001	0.1	0.05
		"Best"	1E-2	0	2	0.01	1	0.25	9E-3	0	1	0.01	0.5	0.2	1E-3	0	1	0.01	0.5	0.1
		High	1E-1	0	8	1	10	20	9E-2	0	3	1	10	20	1E-2	0	5	1	10	20
4	Corrosion (for example, corrosion near water interface in bilges, corrosion of expansion bellows, corrosion of pipe caps, and so on)	Low	1E-4	0	0.4	0.01	0.01	0.05	9E-5	0	0.3	0.01	0.001	0.05	1E-5	0	0.1	0.01	0.001	0.05
		"Best"	1E-3	0	4	0.1	0.05	0.2	9E-4	0	8	0.05	0.025	0.1	1E-4	0	1	0.05	0.025	0.1
		High	1E-2	0	15	5	1	15	9E-3	0	12	2	1	15	1E-3	0	3	2	10	15
5	Fatigue failures (for example, bellows fatigue failure)	Low	1E-4	0	0.2	1E-3	0.001	1E-3	9E-5	0	0.1	0.001	0.001	9E-5	1E-5	0	0.1	0.001	1E-4	1E-4
		"Best"	1E-3	0	0.5	0.01	0.01	1E-3	9E-4	0	0.2	0.01	0.01	9E-4	1E-4	0	0.3	0.01	0.001	1E-4
		High	1E-2	0	3	5	0.1	8	9E-3	0	1	0.5	0.1	8	1E-3	0	2	0.5	0.01	8
	TOTALS	Low	0	0	1.9	0.02	0.31	0.25	0	0	1	0.02	0.21	0.2	0	0	0.9	0.02	0.21	0.16
		"Best"	0.02	0	8.5	0.23	2.31	1.05	0.02	0	10.3	0.13	1.13	0.65	0	0	3.2	0.13	1.18	0.5
		High	0.23	0	39	17	25.1	67	0.21	0	22	6.5	24.1	70	0.02	0	20	6.5	32	70

F
EXPERT ELICITATION RESULTS

This appendix presents the detailed results of the statistical analysis of the expert elicitation.

The input datum used was the trim mean. That is, the lowest and highest experts were not included in the development of the community distribution. This treatment was performed for several reasons. One expert used zero several times in the assignment of the probability of ILRT failure. Zeros are difficult to treat in the statistical evaluation of the expert input. Therefore, this expert was not included in the development of the community distribution. Since the lowest expert was not included in the development of the community distribution, it was prudent to not include the highest expert in the development of the community distribution as well. This treatment results in the use of a four-expert set as opposed to the six to develop the community distribution, and therefore the community distribution represents the center of the input data collected.

In addition, no community distribution was developed for the small containment case. This is a result of the fact that analysis of the small containment input data actually produces similar values for the probability of a leakage pathway in the small containments. The differences are very small and do not represent a significant difference in the probability; therefore, the small containment case was not evaluated. It should be noted that one expert did not complete small containment input sheets since he believed that there was no reason to treat the small containments differently than the large containment type.

Both of the above treatments of the input data were discussed with experts during the elicitation meeting as being potential treatments of the final results. Experts agreed with this treatment.

The following tables and figures present the results of the expert elicitation process. The following tables are presented:

- Table F-1: Large Containment – Construction Error or Deficiency
- Table F-2: Large Containment – Human Error (Testing or Maintenance)
- Table F-3: Large Containment – Human Error (Design Error)
- Table F-4: Large Containment – Corrosion
- Table F-5: Large Containment – Fatigue Failures
- Table F-6: Large Containment – All Failure Modes
- Table F-7: Small Containment – Construction Error or Deficiency
- Table F-8: Small Containment – Human Error (Testing or Maintenance)

- Table F-9: Small Containment – Human Error (Design Error)
- Table F-10: Small Containment – Corrosion
- Table F-11: Small Containment – Fatigue Failures
- Table F-12: Small Containment – All Failure Modes

Several figures are produced from the tables above. These figures are:

- Figure F-1: Large Containment – All Failure Modes
- Figure F-2: Small Containment – All Failure Modes
- Figure F-3: Comparison of Small and Large Containment – Failure Probability

Table F-1
Large Containment – Construction Error or Deficiency

La	Aggregate			Expert A			Expert D			Expert E			Expert F		
	Mean	Lower	Upper	Mean	Lower	Upper	Mean	Lower	Upper	Mean	Lower	Upper	Mean	Lower	Upper
1	4.25E-03	3.04E-04	3.36E-02	1.06E-03	1.68E-04	6.62E-03	6.19E-03	1.56E-03	2.42E-02	4.62E-03	8.90E-04	2.36E-02	5.13E-03	2.28E-04	1.04E-01
2	2.89E-03	1.07E-04	4.11E-02	4.01E-04	6.78E-05	2.37E-03	3.99E-03	9.25E-04	1.70E-02	3.16E-03	5.66E-04	1.75E-02	4.01E-03	4.58E-05	2.62E-01
5	1.72E-03	2.01E-05	8.25E-02	8.99E-05	1.53E-05	5.28E-04	2.10E-03	2.74E-04	1.60E-02	1.84E-03	2.17E-04	1.55E-02	2.85E-03	3.38E-06	7.08E-01
10	1.15E-03	4.52E-06	1.61E-01	2.40E-05	3.48E-06	1.66E-04	1.24E-03	7.67E-05	1.95E-02	1.19E-03	8.04E-05	1.72E-02	2.17E-03	3.57E-07	9.30E-01
20	7.68E-04	6.97E-07	3.18E-01	5.33E-06	5.12E-07	5.56E-05	6.93E-04	1.65E-05	2.82E-02	7.39E-04	2.44E-05	2.19E-02	1.63E-03	3.03E-08	9.89E-01
35	5.49E-04	1.18E-07	4.99E-01	1.36E-06	7.46E-08	2.46E-05	4.19E-04	3.99E-06	4.22E-02	4.92E-04	8.10E-06	2.91E-02	1.29E-03	3.53E-09	9.98E-01
50	4.43E-04	3.30E-08	6.31E-01	5.23E-07	1.80E-08	1.52E-05	2.98E-04	1.47E-06	5.70E-02	3.76E-04	3.77E-06	3.61E-02	1.10E-03	8.30E-10	9.99E-01
100	2.91E-04	1.91E-09	8.46E-01	6.75E-08	7.20E-10	6.32E-06	1.48E-04	1.74E-07	1.12E-01	2.16E-04	7.38E-07	5.95E-02	8.00E-04	4.18E-11	1.00E+00
200	1.91E-04	6.49E-11	9.54E-01	6.52E-09	1.48E-11	2.87E-06	6.90E-05	1.53E-08	2.38E-01	1.19E-04	1.19E-07	1.07E-01	5.75E-04	1.65E-12	1.00E+00
500	1.09E-04	3.37E-13	9.92E-01	1.77E-10	2.66E-14	1.18E-06	2.28E-05	3.76E-10	5.80E-01	5.12E-05	7.75E-09	2.53E-01	3.62E-04	1.56E-14	1.00E+00
600	9.75E-05	1.12E-13	9.95E-01	7.99E-11	6.31E-15	1.01E-06	1.80E-05	1.67E-10	6.59E-01	4.29E-05	4.29E-09	3.00E-01	3.29E-04	5.82E-15	1.00E+00
1000	7.13E-05	2.99E-15	9.99E-01	7.37E-12	7.82E-17	6.95E-07	9.04E-06	1.50E-11	8.45E-01	2.56E-05	7.55E-10	4.65E-01	2.51E-04	3.32E-16	1.00E+00
2000	4.65E-05	1.30E-17	1.00E+00	1.96E-13	7.95E-20	4.82E-07	3.31E-06	4.00E-13	9.65E-01	1.22E-05	5.75E-11	7.21E-01	1.70E-04	5.28E-18	1.00E+00
5000	2.62E-05	3.00E-21	1.00E+00	7.26E-16	1.26E-24	4.16E-07	7.69E-07	1.66E-15	9.97E-01	4.22E-06	1.27E-12	9.33E-01	9.97E-05	1.37E-20	1.00E+00
10000	1.67E-05	1.46E-24	1.00E+00	5.22E-18	5.21E-29	5.23E-07	2.28E-07	1.46E-17	1.00E+00	1.77E-06	5.02E-14	9.84E-01	6.50E-05	1.03E-22	1.00E+00

Table F-2
Large Containment – Human Error (Testing or Maintenance)

La	Aggregate			Expert A			Expert D			Expert E			Expert F		
	Mean	Lower	Upper	Mean	Lower	Upper	Mean	Lower	Upper	Mean	Lower	Upper	Mean	Lower	Upper
1	7.90E-03	1.23E-03	3.47E-02	4.27E-03	1.68E-03	1.08E-02	6.10E-03	8.31E-04	4.33E-02	1.45E-02	6.53E-03	3.21E-02	6.69E-03	8.19E-04	5.25E-02
2	4.48E-03	2.23E-04	7.08E-02	1.58E-03	6.10E-04	4.09E-03	4.40E-03	3.67E-04	5.06E-02	9.04E-03	1.04E-03	7.38E-02	2.88E-03	3.05E-05	2.15E-01
5	2.07E-03	4.83E-06	3.18E-01	3.20E-04	1.02E-04	9.97E-04	2.77E-03	6.82E-05	1.02E-01	4.43E-03	3.81E-05	3.42E-01	7.60E-04	5.56E-08	9.12E-01
10	1.15E-03	8.34E-08	7.63E-01	7.37E-05	1.66E-05	3.28E-04	1.90E-03	1.37E-05	2.09E-01	2.41E-03	1.74E-06	7.71E-01	2.27E-04	9.26E-11	9.98E-01
20	6.42E-04	5.74E-10	9.79E-01	1.30E-05	1.62E-06	1.05E-04	1.28E-03	2.17E-06	4.29E-01	1.22E-03	4.46E-08	9.71E-01	5.55E-05	2.76E-14	1.00E+00
35	3.99E-04	4.08E-12	9.98E-01	2.56E-06	1.62E-07	4.04E-05	9.07E-04	4.13E-07	6.66E-01	6.71E-04	1.44E-09	9.97E-01	1.50E-05	8.68E-18	1.00E+00
50	2.93E-04	1.36E-13	1.00E+00	8.02E-07	2.97E-08	2.16E-05	7.23E-04	1.32E-07	7.98E-01	4.44E-04	1.25E-10	9.99E-01	5.94E-06	2.23E-20	1.00E+00
100	1.61E-04	3.23E-17	1.00E+00	6.21E-08	6.26E-10	6.17E-06	4.56E-04	1.20E-08	9.45E-01	1.86E-04	5.89E-13	1.00E+00	7.85E-07	2.43E-26	1.00E+00
200	8.75E-05	2.10E-21	1.00E+00	3.02E-09	5.47E-12	1.67E-06	2.79E-04	8.43E-10	9.89E-01	7.10E-05	1.10E-15	1.00E+00	7.39E-08	9.32E-34	1.00E+00
500	3.89E-05	6.28E-29	1.00E+00	2.34E-11	1.98E-15	2.77E-07	1.39E-04	1.63E-11	9.99E-01	1.67E-05	5.24E-20	1.00E+00	1.75E-09	2.41E-46	1.00E+00
600	3.30E-05	1.37E-30	1.00E+00	7.78E-12	3.13E-16	1.93E-07	1.20E-04	7.00E-12	1.00E+00	1.22E-05	5.61E-21	1.00E+00	7.54E-10	2.70E-49	1.00E+00
1000	2.09E-05	1.01E-35	1.00E+00	2.70E-13	1.04E-18	7.01E-08	7.87E-05	5.81E-13	1.00E+00	4.86E-06	6.55E-24	1.00E+00	5.90E-11	1.75E-58	1.00E+00
2000	1.11E-05	5.50E-44	1.00E+00	1.38E-15	1.06E-22	1.81E-08	4.30E-05	1.48E-14	1.00E+00	1.23E-06	1.90E-28	1.00E+00	1.12E-12	2.04E-73	1.00E+00
5000	4.61E-06	1.09E-57	1.00E+00	2.89E-19	2.45E-29	3.41E-09	1.83E-05	6.65E-17	1.00E+00	1.57E-07	1.40E-35	1.00E+00	2.12E-15	1.78E-98	1.00E+00
10000	2.28E-06	2.92E-72	1.00E+00	1.21E-22	1.25E-35	1.16E-09	9.10E-06	7.02E-19	1.00E+00	2.69E-08	5.69E-42	1.00E+00	7.23E-18	2.09E-122	1.00E+00

Table F-3
Large Containment – Human Error (Design Error)

La	Aggregate			Expert A			Expert D			Expert E			Expert F		
	Mean	Lower	Upper	Mean	Lower	Upper	Mean	Lower	Upper	Mean	Lower	Upper	Mean	Lower	Upper
1	1.07E-02	3.18E-03	3.06E-02	7.43E-03	5.46E-03	1.01E-02	1.29E-02	6.03E-03	2.74E-02	1.45E-02	6.53E-03	3.21E-02	8.01E-03	1.33E-03	4.68E-02
2	5.65E-03	5.26E-04	6.79E-02	2.72E-03	1.81E-03	4.09E-03	5.83E-03	1.62E-03	2.08E-02	9.04E-03	1.04E-03	7.38E-02	5.03E-03	8.63E-05	2.29E-01
5	2.28E-03	1.78E-05	3.38E-01	5.22E-04	2.33E-04	1.17E-03	1.62E-03	8.99E-05	2.85E-02	4.43E-03	3.81E-05	3.42E-01	2.54E-03	5.26E-07	9.25E-01
10	1.11E-03	6.50E-07	7.87E-01	1.11E-04	3.02E-05	4.07E-04	5.00E-04	4.52E-06	5.25E-02	2.41E-03	1.74E-06	7.71E-01	1.43E-03	4.68E-09	9.98E-01
20	5.32E-04	1.37E-08	9.76E-01	1.71E-05	2.33E-06	1.26E-04	1.24E-04	9.96E-08	1.35E-01	1.22E-03	4.46E-08	9.71E-01	7.61E-04	1.86E-11	1.00E+00
35	2.86E-04	2.84E-10	9.97E-01	2.88E-06	1.88E-07	4.42E-05	3.37E-05	2.22E-09	3.39E-01	6.71E-04	1.44E-09	9.97E-01	4.38E-04	1.12E-13	1.00E+00
50	1.90E-04	2.02E-11	1.00E+00	7.97E-07	2.95E-08	2.16E-05	1.33E-05	1.32E-10	5.72E-01	4.44E-04	1.25E-10	9.99E-01	3.01E-04	3.06E-15	1.00E+00
100	8.15E-05	4.24E-14	1.00E+00	4.48E-08	4.23E-10	4.75E-06	1.70E-06	1.95E-13	9.37E-01	1.86E-04	5.89E-13	1.00E+00	1.38E-04	1.24E-18	1.00E+00
200	3.24E-05	2.63E-17	1.00E+00	1.40E-09	2.19E-12	8.93E-07	1.50E-07	5.62E-17	9.98E-01	7.10E-05	1.10E-15	1.00E+00	5.86E-05	1.52E-22	1.00E+00
500	8.34E-06	1.89E-22	1.00E+00	4.68E-12	2.88E-16	7.59E-08	3.00E-09	4.98E-23	1.00E+00	1.67E-05	5.24E-20	1.00E+00	1.66E-05	1.24E-28	1.00E+00
600	6.24E-06	1.40E-23	1.00E+00	1.26E-12	3.54E-17	4.49E-08	1.24E-09	1.87E-24	1.00E+00	1.22E-05	5.61E-21	1.00E+00	1.27E-05	5.52E-30	1.00E+00
1000	2.66E-06	3.02E-27	1.00E+00	2.23E-14	5.09E-20	9.74E-09	8.28E-11	6.53E-29	1.00E+00	4.86E-06	6.55E-24	1.00E+00	5.76E-06	4.90E-34	1.00E+00
2000	7.58E-07	6.30E-33	1.00E+00	3.54E-17	1.17E-24	1.08E-09	1.18E-12	3.34E-36	1.00E+00	1.23E-06	1.90E-28	1.00E+00	1.80E-06	3.20E-40	1.00E+00
5000	1.21E-07	3.21E-42	1.00E+00	8.93E-22	1.64E-32	4.87E-11	1.28E-15	1.50E-48	1.00E+00	1.57E-07	1.40E-35	1.00E+00	3.27E-07	9.05E-50	1.00E+00
10000	2.61E-08	6.55E-51	1.00E+00	4.32E-26	4.34E-40	4.31E-12	2.40E-18	1.86E-60	1.00E+00	2.69E-08	5.69E-42	1.00E+00	7.76E-08	3.52E-58	1.00E+00

Table F-4
Large Containment – Corrosion

La	Aggregate			Expert A			Expert D			Expert E			Expert F		
	Mean	Lower	Upper	Mean	Lower	Upper	Mean	Lower	Upper	Mean	Lower	Upper	Mean	Lower	Upper
1	4.30E-03	1.76E-04	4.03E-02	5.54E-04	1.22E-04	2.50E-03	8.32E-03	2.70E-03	2.54E-02	3.57E-03	1.06E-03	1.19E-02	4.76E-03	7.92E-05	2.24E-01
2	2.65E-03	7.48E-05	4.54E-02	1.99E-04	5.89E-05	6.70E-04	4.97E-03	1.38E-03	1.77E-02	2.03E-03	6.48E-04	6.32E-03	3.42E-03	2.08E-05	3.60E-01
5	1.34E-03	2.06E-05	8.81E-02	4.12E-05	2.03E-05	8.33E-05	2.31E-03	3.18E-04	1.66E-02	8.80E-04	3.08E-04	2.51E-03	2.14E-03	1.33E-06	7.74E-01
10	7.76E-04	6.00E-06	1.78E-01	1.03E-05	6.52E-06	1.64E-05	1.20E-03	7.13E-05	1.99E-02	4.35E-04	1.58E-04	1.20E-03	1.46E-03	9.53E-08	9.57E-01
20	4.40E-04	7.08E-07	3.46E-01	2.15E-06	8.08E-07	5.72E-06	5.83E-04	1.18E-05	2.82E-02	2.00E-04	7.08E-05	5.67E-04	9.73E-04	4.50E-09	9.95E-01
35	2.75E-04	8.39E-08	5.81E-01	5.18E-07	9.24E-08	2.90E-06	3.08E-04	2.17E-06	4.18E-02	1.01E-04	3.24E-05	3.16E-04	6.89E-04	2.86E-10	9.99E-01
50	2.03E-04	1.87E-08	7.20E-01	1.93E-07	1.94E-08	1.92E-06	1.99E-04	6.57E-07	5.67E-02	6.37E-05	1.83E-05	2.22E-04	5.48E-04	4.29E-11	1.00E+00
100	1.12E-04	6.65E-10	9.19E-01	2.33E-08	6.08E-10	8.92E-07	7.95E-05	4.87E-08	1.15E-01	2.42E-05	5.10E-06	1.15E-04	3.44E-04	7.86E-13	1.00E+00
200	6.18E-05	1.54E-11	9.84E-01	2.10E-09	1.02E-11	4.35E-07	2.88E-05	2.39E-09	2.58E-01	8.34E-06	1.13E-06	6.17E-05	2.10E-04	9.29E-15	1.00E+00
500	2.81E-05	4.68E-14	9.98E-01	5.25E-11	1.46E-14	1.89E-07	6.36E-06	2.15E-11	6.53E-01	1.73E-06	1.06E-07	2.82E-05	1.04E-04	1.27E-17	1.00E+00
600	2.40E-05	1.08E-14	9.99E-01	2.33E-11	3.32E-15	1.63E-07	4.59E-06	7.55E-12	7.36E-01	1.23E-06	6.29E-08	2.42E-05	9.01E-05	3.09E-18	1.00E+00
1000	1.53E-05	2.21E-16	1.00E+00	2.06E-12	3.72E-17	1.14E-07	1.76E-06	3.24E-13	9.05E-01	4.58E-07	1.32E-08	1.59E-05	5.91E-05	4.79E-20	1.00E+00
2000	8.21E-06	4.92E-19	1.00E+00	5.19E-14	3.35E-20	8.04E-08	4.22E-07	2.62E-15	9.86E-01	1.06E-07	1.23E-09	9.12E-06	3.23E-05	1.03E-22	1.00E+00
5000	3.45E-06	5.05E-23	1.00E+00	1.82E-16	4.74E-25	7.01E-08	5.06E-08	1.49E-18	9.99E-01	1.22E-08	3.33E-11	4.49E-06	1.37E-05	1.22E-26	1.00E+00
10000	1.72E-06	1.09E-26	1.00E+00	1.28E-18	1.87E-29	8.79E-08	8.32E-09	1.99E-21	1.00E+00	1.97E-09	1.43E-12	2.71E-06	6.87E-06	6.01E-30	1.00E+00

Table F-5
Large Containment – Fatigue Failures

La	Aggregate			Expert A			Expert D			Expert E			Expert F		
	Mean	Lower	Upper	Mean	Lower	Upper	Mean	Lower	Upper	Mean	Lower	Upper	Mean	Lower	Upper
1	8.97E-05	7.88E-36	1.00E+00	4.77E-05	1.97E-06	1.15E-03	2.36E-04	7.82E-07	6.67E-02	1.58E-05	1.03E-06	2.43E-04	5.91E-05	3.15E-65	1.00E+00
2	5.37E-05	2.26E-207	1.00E+00	2.25E-05	1.23E-06	4.14E-04	1.70E-04	5.62E-07	4.87E-02	1.09E-05	9.20E-07	1.30E-04	1.18E-05	0.00E+00	0.00E+00
5	3.06E-05	0.00E+00	0.00E+00	7.66E-06	6.58E-07	8.92E-05	1.07E-04	1.09E-07	9.55E-02	6.59E-06	6.00E-07	7.23E-05	9.29E-07	0.00E+00	0.00E+00
10	2.05E-05	0.00E+00	0.00E+00	3.15E-06	4.13E-07	2.41E-05	7.44E-05	1.43E-08	2.79E-01	4.42E-06	3.29E-07	5.94E-05	9.34E-08	0.00E+00	0.00E+00
20	1.38E-05	0.00E+00	0.00E+00	1.21E-06	2.60E-07	5.67E-06	5.10E-05	1.16E-09	6.92E-01	2.93E-06	1.42E-07	6.02E-05	6.43E-09	0.00E+00	0.00E+00
35	9.94E-06	0.00E+00	0.00E+00	5.34E-07	1.78E-07	1.60E-06	3.71E-05	1.14E-10	9.24E-01	2.08E-06	6.27E-08	6.90E-05	5.39E-10	0.00E+00	0.00E+00
50	8.04E-06	0.00E+00	0.00E+00	3.08E-07	1.38E-07	6.88E-07	3.02E-05	2.32E-11	9.75E-01	1.66E-06	3.51E-08	7.88E-05	9.37E-11	0.00E+00	0.00E+00
100	5.28E-06	0.00E+00	0.00E+00	9.95E-08	6.23E-08	1.59E-07	2.00E-05	8.26E-13	9.98E-01	1.07E-06	1.02E-08	1.11E-04	2.06E-12	0.00E+00	0.00E+00
200	3.42E-06	0.00E+00	0.00E+00	2.95E-08	1.06E-08	8.26E-08	1.30E-05	2.18E-14	1.00E+00	6.73E-07	2.63E-09	1.72E-04	2.41E-14	0.00E+00	0.00E+00
500	1.88E-06	0.00E+00	0.00E+00	5.15E-09	5.26E-10	5.03E-08	7.15E-06	1.13E-16	1.00E+00	3.58E-07	3.71E-10	3.46E-04	2.13E-17	0.00E+00	0.00E+00
600	1.66E-06	0.00E+00	0.00E+00	3.56E-09	2.74E-10	4.63E-08	6.33E-06	3.71E-17	1.00E+00	3.15E-07	2.46E-10	4.03E-04	4.41E-18	0.00E+00	0.00E+00
1000	1.17E-06	0.00E+00	0.00E+00	1.22E-09	3.97E-11	3.76E-08	4.46E-06	1.48E-18	1.00E+00	2.18E-07	7.47E-11	6.37E-04	3.74E-20	0.00E+00	0.00E+00
2000	7.16E-07	0.00E+00	0.00E+00	2.60E-10	2.27E-12	2.99E-08	2.73E-06	1.42E-20	1.00E+00	1.31E-07	1.36E-11	1.25E-03	2.29E-23	0.00E+00	0.00E+00
5000	3.63E-07	0.00E+00	0.00E+00	2.82E-11	3.24E-14	2.45E-08	1.39E-06	1.84E-23	1.00E+00	6.46E-08	1.23E-12	3.40E-03	1.93E-28	0.00E+00	0.00E+00
10000	2.12E-07	0.00E+00	0.00E+00	4.52E-12	8.83E-16	2.31E-08	8.10E-07	8.12E-26	1.00E+00	3.72E-08	1.76E-13	7.79E-03	5.04E-33	0.00E+00	0.00E+00

Table F-6
Large Containment – All Failure Modes

La	Aggregate			Expert A			Expert D			Expert E			Expert F		
	Mean	Lower	Upper	Mean	Lower	Upper	Mean	Lower	Upper	Mean	Lower	Upper	Mean	Lower	Upper
1	2.65E-02	4.50E-03	1.33E-01	1.27E-02	4.99E-03	3.20E-02	3.06E-02	7.94E-03	1.11E-01	3.68E-02	1.41E-02	9.27E-02	2.57E-02	1.43E-03	3.27E-01
2	1.59E-02	1.23E-03	1.99E-01	5.24E-03	2.00E-03	1.36E-02	1.87E-02	3.71E-03	8.90E-02	2.30E-02	4.00E-03	1.22E-01	1.65E-02	1.29E-04	6.86E-01
5	7.42E-03	7.54E-05	5.35E-01	1.22E-03	4.21E-04	3.55E-03	8.77E-03	6.51E-04	1.07E-01	1.12E-02	3.18E-04	2.86E-01	8.51E-03	1.23E-06	9.84E-01
10	3.88E-03	4.71E-06	8.57E-01	3.13E-04	8.69E-05	1.13E-03	4.49E-03	1.02E-04	1.66E-01	5.90E-03	2.61E-05	5.74E-01	4.79E-03	1.32E-08	9.99E-01
20	1.88E-03	1.38E-07	9.82E-01	6.08E-05	1.12E-05	3.30E-04	2.10E-03	9.95E-06	3.07E-01	2.85E-03	1.21E-06	8.71E-01	2.52E-03	5.44E-11	1.00E+00
35	9.86E-04	5.56E-09	9.98E-01	1.27E-05	1.41E-06	1.15E-04	1.05E-03	1.03E-06	5.16E-01	1.47E-03	6.20E-08	9.72E-01	1.42E-03	2.90E-13	1.00E+00
50	6.33E-04	5.34E-10	9.99E-01	4.13E-06	2.98E-07	5.73E-05	6.46E-04	1.99E-07	6.78E-01	9.26E-04	7.19E-09	9.92E-01	9.55E-04	6.79E-15	1.00E+00
100	2.47E-04	2.65E-12	1.00E+00	3.33E-07	8.09E-09	1.37E-05	2.29E-04	4.88E-09	9.15E-01	3.44E-04	5.64E-11	1.00E+00	4.14E-04	1.63E-18	1.00E+00
200	8.57E-05	4.17E-15	1.00E+00	1.61E-08	8.76E-11	2.96E-06	7.01E-05	5.59E-11	9.89E-01	1.11E-04	1.61E-13	1.00E+00	1.62E-04	8.21E-23	1.00E+00
500	1.75E-05	1.68E-19	1.00E+00	1.12E-10	3.76E-14	3.33E-07	1.13E-05	3.71E-14	1.00E+00	1.93E-05	1.08E-17	1.00E+00	3.93E-05	1.02E-29	1.00E+00
600	1.24E-05	1.61E-20	1.00E+00	3.57E-11	6.03E-15	2.11E-07	7.57E-06	6.94E-15	1.00E+00	1.31E-05	1.19E-18	1.00E+00	2.89E-05	2.80E-31	1.00E+00
1000	4.50E-06	1.28E-23	1.00E+00	1.06E-12	1.98E-17	5.74E-08	2.27E-06	4.08E-17	1.00E+00	4.14E-06	1.36E-21	1.00E+00	1.16E-05	4.97E-36	1.00E+00
2000	1.01E-06	1.74E-28	1.00E+00	3.94E-15	1.67E-21	9.27E-09	3.62E-07	1.20E-20	1.00E+00	7.13E-07	2.96E-26	1.00E+00	2.95E-06	1.90E-43	1.00E+00
5000	1.11E-07	2.69E-36	1.00E+00	4.04E-19	2.01E-28	8.11E-10	2.14E-08	2.26E-26	1.00E+00	4.73E-08	7.77E-34	1.00E+00	3.75E-07	3.02E-55	1.00E+00
10000	1.73E-08	1.92E-43	1.00E+00	7.40E-23	3.99E-35	1.37E-10	1.77E-09	1.17E-31	1.00E+00	4.31E-09	7.43E-41	1.00E+00	6.32E-08	6.09E-66	1.00E+00

Table F-7
Small Containment – Construction Error or Deficiency

La	Aggregate			Expert A			Expert D			Expert E			Expert F		
	Mean	Lower	Upper	Mean	Lower	Upper	Mean	Lower	Upper	Mean	Lower	Upper	Mean	Lower	Upper
1	3.55E-03	1.91E-04	3.74E-02	1.06E-03	1.68E-04	6.62E-03	6.19E-03	1.56E-03	2.42E-02	1.83E-03	1.00E-04	3.24E-02	5.13E-03	2.28E-04	1.04E-01
2	2.43E-03	8.23E-05	4.60E-02	4.01E-04	6.78E-05	2.37E-03	3.99E-03	9.25E-04	1.70E-02	1.31E-03	7.87E-05	2.14E-02	4.01E-03	4.58E-05	2.62E-01
5	1.47E-03	1.88E-05	9.31E-02	8.99E-05	1.53E-05	5.28E-04	2.10E-03	2.74E-04	1.60E-02	8.19E-04	4.05E-05	1.63E-02	2.85E-03	3.38E-06	7.08E-01
10	9.98E-04	4.24E-06	1.61E-01	2.40E-05	3.48E-06	1.66E-04	1.24E-03	7.67E-05	1.95E-02	5.62E-04	1.81E-05	1.72E-02	2.17E-03	3.57E-07	9.30E-01
20	6.77E-04	6.74E-07	3.18E-01	5.33E-06	5.12E-07	5.56E-05	6.93E-04	1.65E-05	2.82E-02	3.78E-04	6.26E-06	2.23E-02	1.63E-03	3.03E-08	9.89E-01
35	4.94E-04	1.12E-07	5.32E-01	1.36E-06	7.46E-08	2.46E-05	4.19E-04	3.99E-06	4.22E-02	2.70E-04	2.26E-06	3.13E-02	1.29E-03	3.53E-09	9.98E-01
50	4.03E-04	3.11E-08	6.63E-01	5.23E-07	1.80E-08	1.52E-05	2.98E-04	1.47E-06	5.70E-02	2.17E-04	1.10E-06	4.10E-02	1.10E-03	8.30E-10	9.99E-01
100	2.72E-04	1.82E-09	8.46E-01	6.75E-08	7.20E-10	6.32E-06	1.48E-04	1.74E-07	1.12E-01	1.38E-04	2.33E-07	7.59E-02	8.00E-04	4.18E-11	1.00E+00
200	1.83E-04	6.29E-11	9.54E-01	6.52E-09	1.48E-11	2.87E-06	6.90E-05	1.53E-08	2.38E-01	8.64E-05	4.12E-08	1.54E-01	5.75E-04	1.65E-12	1.00E+00
500	1.07E-04	3.30E-13	9.94E-01	1.77E-10	2.66E-14	1.18E-06	2.28E-05	3.76E-10	5.80E-01	4.46E-05	3.13E-09	3.89E-01	3.62E-04	1.56E-14	1.00E+00
600	9.65E-05	1.02E-13	9.96E-01	7.99E-11	6.31E-15	1.01E-06	1.80E-05	1.67E-10	6.59E-01	3.89E-05	1.80E-09	4.57E-01	3.29E-04	5.82E-15	1.00E+00
1000	7.15E-05	2.85E-15	9.99E-01	7.37E-12	7.82E-17	6.95E-07	9.04E-06	1.50E-11	8.45E-01	2.62E-05	3.57E-10	6.58E-01	2.51E-04	3.32E-16	1.00E+00
2000	4.72E-05	1.33E-17	1.00E+00	1.96E-13	7.95E-20	4.82E-07	3.31E-06	4.00E-13	9.65E-01	1.50E-05	3.35E-11	8.70E-01	1.70E-04	5.28E-18	1.00E+00
5000	2.68E-05	2.86E-21	1.00E+00	7.26E-16	1.26E-24	4.16E-07	7.69E-07	1.66E-15	9.97E-01	6.84E-06	1.06E-12	9.78E-01	9.97E-05	1.37E-20	1.00E+00
10000	1.72E-05	1.55E-24	1.00E+00	5.22E-18	5.21E-29	5.23E-07	2.28E-07	1.46E-17	1.00E+00	3.64E-06	6.05E-14	9.95E-01	6.50E-05	1.03E-22	1.00E+00

Table F-8
Small Containment – Human Error (Testing or Maintenance)

La	Aggregate			Expert A			Expert D			Expert E			Expert F		
	Mean	Lower	Upper	Mean	Lower	Upper	Mean	Lower	Upper	Mean	Lower	Upper	Mean	Lower	Upper
1	5.90E-03	1.16E-04	1.03E-01	5.63E-03	2.59E-03	1.22E-02	6.10E-03	8.31E-04	4.33E-02	9.12E-03	9.75E-04	7.99E-02	2.76E-03	6.26E-06	5.51E-01
2	3.87E-03	2.72E-06	5.93E-01	2.96E-03	1.35E-03	6.51E-03	4.40E-03	3.67E-04	5.06E-02	6.22E-03	5.91E-04	6.21E-02	1.91E-03	5.48E-09	9.98E-01
5	2.15E-03	1.03E-09	9.99E-01	1.12E-03	4.38E-04	2.87E-03	2.77E-03	6.82E-05	1.02E-01	3.57E-03	1.71E-04	6.98E-02	1.13E-03	2.40E-15	1.00E+00
10	1.34E-03	8.57E-13	1.00E+00	4.84E-04	1.47E-04	1.59E-03	1.90E-03	1.37E-05	2.09E-01	2.25E-03	4.37E-05	1.04E-01	7.33E-04	4.99E-21	1.00E+00
20	8.24E-04	6.27E-16	1.00E+00	1.88E-04	3.92E-05	9.02E-04	1.28E-03	2.17E-06	4.29E-01	1.37E-03	8.11E-06	1.88E-01	4.65E-04	2.02E-27	1.00E+00
35	5.48E-04	5.67E-19	1.00E+00	8.05E-05	1.13E-05	5.76E-04	9.07E-04	4.13E-07	6.66E-01	8.88E-04	1.67E-06	3.22E-01	3.16E-04	3.96E-33	1.00E+00
50	4.19E-04	4.60E-21	1.00E+00	4.49E-05	4.64E-06	4.35E-04	7.23E-04	1.32E-07	7.98E-01	6.64E-04	5.47E-07	4.47E-01	2.44E-04	4.91E-37	1.00E+00
100	2.45E-04	1.75E-25	1.00E+00	1.30E-05	6.69E-07	2.53E-04	4.56E-04	1.20E-08	9.45E-01	3.65E-04	4.95E-08	7.30E-01	1.44E-04	2.89E-45	1.00E+00
200	1.39E-04	2.07E-30	1.00E+00	3.23E-06	7.03E-08	1.49E-04	2.79E-04	8.43E-10	9.89E-01	1.91E-04	3.20E-09	9.20E-01	8.27E-05	2.14E-54	1.00E+00
500	6.29E-05	7.16E-37	1.00E+00	3.92E-07	2.05E-09	7.52E-05	1.39E-04	1.63E-11	9.99E-01	7.50E-05	4.88E-11	9.91E-01	3.75E-05	5.47E-68	1.00E+00
600	5.34E-05	2.00E-38	1.00E+00	2.48E-07	9.30E-10	6.59E-05	1.20E-04	7.00E-12	1.00E+00	6.15E-05	1.95E-11	9.95E-01	3.18E-05	6.49E-71	1.00E+00
1000	3.33E-05	5.19E-43	1.00E+00	6.32E-08	8.65E-11	4.61E-05	7.87E-05	5.81E-13	1.00E+00	3.45E-05	1.29E-12	9.99E-01	1.97E-05	1.57E-79	1.00E+00
2000	1.70E-05	8.01E-50	1.00E+00	8.11E-09	2.25E-12	2.93E-05	4.30E-05	1.48E-14	1.00E+00	1.49E-05	2.16E-14	1.00E+00	9.97E-06	2.90E-92	1.00E+00
5000	6.62E-06	5.75E-60	1.00E+00	3.62E-10	7.57E-15	1.73E-05	1.83E-05	6.65E-17	1.00E+00	4.42E-06	4.51E-17	1.00E+00	3.79E-06	4.21E-111	1.00E+00
10000	3.11E-06	1.35E-68	1.00E+00	2.46E-11	4.76E-17	1.27E-05	9.10E-06	7.02E-19	1.00E+00	1.62E-06	2.20E-19	1.00E+00	1.72E-06	5.14E-127	1.00E+00

Table F-9
Small Containment – Human Error (Design Error)

La	Aggregate			Expert A			Expert D			Expert E			Expert F		
	Mean	Lower	Upper	Mean	Lower	Upper	Mean	Lower	Upper	Mean	Lower	Upper	Mean	Lower	Upper
1	1.16E-02	2.81E-03	4.33E-02	9.16E-03	6.69E-03	1.25E-02	1.29E-02	6.03E-03	2.74E-02	1.57E-02	3.80E-03	6.25E-02	8.44E-03	1.18E-03	5.80E-02
2	6.49E-03	4.93E-04	8.25E-02	4.00E-03	2.51E-03	6.35E-03	5.83E-03	1.62E-03	2.08E-02	1.01E-02	1.88E-03	5.26E-02	6.01E-03	4.55E-05	4.46E-01
5	2.91E-03	2.02E-05	3.89E-01	1.06E-03	4.31E-04	2.62E-03	1.62E-03	8.99E-05	2.85E-02	5.27E-03	3.93E-04	6.66E-02	3.70E-03	1.70E-07	9.88E-01
10	1.58E-03	8.79E-07	8.05E-01	3.17E-04	7.77E-05	1.29E-03	5.00E-04	4.52E-06	5.25E-02	3.02E-03	7.94E-05	1.04E-01	2.49E-03	1.22E-09	1.00E+00
20	8.65E-04	2.29E-08	9.80E-01	7.63E-05	9.57E-06	6.07E-04	1.24E-04	9.96E-08	1.35E-01	1.64E-03	1.15E-05	1.90E-01	1.62E-03	4.59E-12	1.00E+00
35	5.33E-04	7.29E-10	9.97E-01	2.01E-05	1.27E-06	3.19E-04	3.37E-05	2.22E-09	3.39E-01	9.51E-04	1.85E-06	3.28E-01	1.13E-03	3.05E-14	1.00E+00
50	3.90E-04	5.71E-11	9.99E-01	7.83E-06	2.94E-07	2.09E-04	1.33E-05	1.32E-10	5.72E-01	6.57E-04	5.11E-07	4.59E-01	8.82E-04	9.70E-16	1.00E+00
100	2.10E-04	2.07E-13	1.00E+00	9.80E-07	1.09E-08	8.85E-05	1.70E-06	1.95E-13	9.37E-01	3.03E-04	3.05E-08	7.50E-01	5.35E-04	6.48E-19	1.00E+00
200	1.11E-04	2.89E-16	1.00E+00	8.50E-08	1.99E-10	3.63E-05	1.50E-07	5.62E-17	9.98E-01	1.28E-04	1.16E-09	9.34E-01	3.14E-04	1.82E-22	1.00E+00
500	4.54E-05	4.31E-21	1.00E+00	1.71E-09	2.69E-13	1.08E-05	3.00E-09	4.98E-23	1.00E+00	3.59E-05	6.87E-12	9.95E-01	1.46E-04	8.30E-28	1.00E+00
600	3.78E-05	3.14E-22	1.00E+00	7.06E-10	5.84E-14	8.53E-06	1.24E-09	1.87E-24	1.00E+00	2.73E-05	2.20E-12	9.97E-01	1.24E-04	5.78E-29	1.00E+00
1000	2.25E-05	1.91E-25	1.00E+00	4.82E-11	5.29E-16	4.38E-06	8.28E-11	6.53E-29	1.00E+00	1.22E-05	7.09E-14	1.00E+00	7.78E-05	2.19E-32	1.00E+00
2000	1.09E-05	8.09E-31	1.00E+00	7.23E-13	2.84E-19	1.84E-06	1.18E-12	3.34E-36	1.00E+00	3.67E-06	3.68E-16	1.00E+00	3.97E-05	1.76E-37	1.00E+00
5000	3.95E-06	4.35E-39	1.00E+00	8.81E-16	1.16E-24	6.69E-07	1.28E-15	1.50E-48	1.00E+00	6.20E-07	1.04E-19	1.00E+00	1.52E-05	4.32E-45	1.00E+00
10000	1.76E-06	6.05E-47	1.00E+00	1.92E-18	1.00E-29	3.70E-07	2.40E-18	1.86E-60	1.00E+00	1.37E-07	7.42E-23	1.00E+00	6.90E-06	1.40E-51	1.00E+00

Table F-10
Small Containment – Corrosion

La	Aggregate			Expert A			Expert D			Expert E			Expert F		
	Mean	Lower	Upper	Mean	Lower	Upper	Mean	Lower	Upper	Mean	Lower	Upper	Mean	Lower	Upper
1	2.99E-03	6.08E-05	3.63E-02	5.73E-04	9.22E-05	3.56E-03	8.32E-03	2.70E-03	2.54E-02	1.11E-03	5.50E-05	2.21E-02	1.96E-03	1.46E-05	2.10E-01
2	1.72E-03	4.62E-05	1.52E-02	2.25E-04	3.91E-05	1.29E-03	4.97E-03	1.38E-03	1.77E-02	6.78E-04	2.19E-05	2.06E-02	1.01E-03	5.91E-05	1.70E-02
5	7.68E-04	6.70E-06	1.96E-02	5.45E-05	9.44E-06	3.14E-04	2.31E-03	3.18E-04	1.66E-02	3.33E-04	1.24E-06	8.25E-02	3.76E-04	6.67E-06	2.07E-02
10	3.91E-04	3.18E-07	8.95E-02	1.59E-05	2.29E-06	1.11E-04	1.20E-03	7.13E-05	1.99E-02	1.86E-04	6.23E-08	3.57E-01	1.62E-04	5.23E-08	3.34E-01
20	1.88E-04	6.55E-09	4.91E-01	3.99E-06	3.71E-07	4.29E-05	5.83E-04	1.18E-05	2.82E-02	9.94E-05	1.84E-09	8.43E-01	6.39E-05	1.13E-10	9.73E-01
35	9.87E-05	1.50E-10	9.08E-01	1.15E-06	6.13E-08	2.16E-05	3.08E-04	2.17E-06	4.18E-02	5.80E-05	7.37E-11	9.79E-01	2.80E-05	3.24E-13	1.00E+00
50	6.39E-05	1.33E-11	9.82E-01	4.89E-07	1.66E-08	1.44E-05	1.99E-04	6.57E-07	5.67E-02	4.05E-05	7.93E-12	9.95E-01	1.60E-05	4.97E-15	1.00E+00
100	2.60E-05	4.74E-14	9.99E-01	7.91E-08	8.88E-10	7.05E-06	7.95E-05	4.87E-08	1.15E-01	1.94E-05	6.87E-14	1.00E+00	4.93E-06	4.96E-19	1.00E+00
200	9.74E-06	5.44E-17	1.00E+00	1.02E-08	2.76E-11	3.77E-06	2.88E-05	2.39E-09	2.58E-01	8.80E-06	3.31E-16	1.00E+00	1.34E-06	9.83E-24	1.00E+00
500	2.35E-06	2.69E-21	1.00E+00	4.56E-10	1.09E-13	1.90E-06	6.36E-06	2.15E-11	6.53E-01	2.83E-06	1.07E-19	1.00E+00	1.93E-07	3.37E-31	1.00E+00
600	1.74E-06	2.32E-22	1.00E+00	2.31E-10	3.15E-14	1.70E-06	4.59E-06	7.55E-12	7.36E-01	2.23E-06	1.87E-20	1.00E+00	1.27E-07	7.10E-33	1.00E+00
1000	7.27E-07	3.42E-25	1.00E+00	3.08E-11	7.32E-16	1.30E-06	1.76E-06	3.24E-13	9.05E-01	1.12E-06	1.07E-22	1.00E+00	3.70E-08	6.07E-38	1.00E+00
2000	2.10E-07	1.36E-29	1.00E+00	1.48E-12	2.16E-18	1.02E-06	4.22E-07	2.62E-15	9.86E-01	4.11E-07	4.90E-26	1.00E+00	5.97E-09	8.80E-46	1.00E+00
5000	3.73E-08	4.35E-37	1.00E+00	1.49E-14	2.27E-22	9.82E-07	5.06E-08	1.49E-18	9.99E-01	9.82E-08	4.96E-31	1.00E+00	3.95E-10	4.30E-58	1.00E+00
10000	9.67E-09	4.91E-43	1.00E+00	2.76E-16	6.09E-26	1.25E-06	8.32E-09	1.99E-21	1.00E+00	3.03E-08	2.70E-35	1.00E+00	3.93E-11	4.11E-69	1.00E+00

Table F-11
Small Containment – Fatigue Failures

La	Aggregate			Expert A			Expert D			Expert E			Expert F		
	Mean	Lower	Upper	Mean	Lower	Upper	Mean	Lower	Upper	Mean	Lower	Upper	Mean	Lower	Upper
1	8.91E-05	8.15E-37	1.00E+00	5.13E-05	2.00E-06	1.31E-03	2.36E-04	7.82E-07	6.67E-02	1.58E-05	1.03E-06	2.43E-04	5.29E-05	3.32E-67	1.00E+00
2	5.18E-05	8.08E-262	1.00E+00	2.30E-05	1.25E-06	4.24E-04	1.70E-04	5.62E-07	4.87E-02	1.09E-05	9.20E-07	1.30E-04	3.52E-06	0.00E+00	0.00E+00
5	3.02E-05	0.00E+00	0.00E+00	7.22E-06	5.72E-07	9.12E-05	1.07E-04	1.09E-07	9.55E-02	6.59E-06	6.00E-07	7.23E-05	3.02E-08	0.00E+00	0.00E+00
10	2.04E-05	0.00E+00	0.00E+00	2.76E-06	2.42E-07	3.16E-05	7.44E-05	1.43E-08	2.79E-01	4.42E-06	3.29E-07	5.94E-05	2.57E-10	0.00E+00	0.00E+00
20	1.37E-05	0.00E+00	0.00E+00	9.78E-07	6.87E-08	1.39E-05	5.10E-05	1.16E-09	6.92E-01	2.93E-06	1.42E-07	6.02E-05	5.90E-13	0.00E+00	0.00E+00
35	9.90E-06	0.00E+00	0.00E+00	3.98E-07	1.76E-08	9.01E-06	3.71E-05	1.14E-10	9.24E-01	2.08E-06	6.27E-08	6.90E-05	1.31E-15	0.00E+00	0.00E+00
50	8.02E-06	0.00E+00	0.00E+00	2.18E-07	6.29E-09	7.53E-06	3.02E-05	2.32E-11	9.75E-01	1.66E-06	3.51E-08	7.88E-05	1.36E-17	0.00E+00	0.00E+00
100	5.27E-06	0.00E+00	0.00E+00	6.27E-08	6.14E-10	6.41E-06	2.00E-05	8.26E-13	9.98E-01	1.07E-06	1.02E-08	1.11E-04	3.11E-22	0.00E+00	0.00E+00
200	3.41E-06	0.00E+00	0.00E+00	1.63E-08	3.99E-11	6.68E-06	1.30E-05	2.18E-14	1.00E+00	6.73E-07	2.63E-09	1.72E-04	3.75E-28	0.00E+00	0.00E+00
500	1.88E-06	0.00E+00	0.00E+00	2.33E-09	5.82E-13	9.32E-06	7.15E-06	1.13E-16	1.00E+00	3.58E-07	3.71E-10	3.46E-04	1.51E-38	0.00E+00	0.00E+00
600	1.66E-06	0.00E+00	0.00E+00	1.54E-09	2.30E-13	1.03E-05	6.33E-06	3.71E-17	1.00E+00	3.15E-07	2.46E-10	4.03E-04	4.79E-41	0.00E+00	0.00E+00
1000	1.17E-06	0.00E+00	0.00E+00	4.64E-10	1.46E-14	1.47E-05	4.46E-06	1.48E-18	1.00E+00	2.18E-07	7.47E-11	6.37E-04	5.87E-49	0.00E+00	0.00E+00
2000	7.16E-07	0.00E+00	0.00E+00	8.12E-11	2.34E-16	2.82E-05	2.73E-06	1.42E-20	1.00E+00	1.31E-07	1.36E-11	1.25E-03	3.12E-62	0.00E+00	0.00E+00
5000	3.63E-07	0.00E+00	0.00E+00	6.51E-12	4.65E-19	9.12E-05	1.39E-06	1.84E-23	1.00E+00	6.46E-08	1.23E-12	3.40E-03	1.51E-85	0.00E+00	0.00E+00
10000	2.12E-07	0.00E+00	0.00E+00	8.06E-13	2.23E-21	2.91E-04	8.10E-07	8.12E-26	1.00E+00	3.72E-08	1.76E-13	7.79E-03	6.84E-109	0.00E+00	0.00E+00

Table F-12
Small Containment – All Failure Modes

La	Aggregate			Expert A			Expert D			Expert E			Expert F		
	Mean	Lower	Upper	Mean	Lower	Upper	Mean	Lower	Upper	Mean	Lower	Upper	Mean	Lower	Upper
1	2.26E-02	2.02E-03	1.86E-01	1.54E-02	6.77E-03	3.46E-02	3.06E-02	7.94E-03	1.11E-01	2.65E-02	3.68E-03	1.67E-01	1.79E-02	3.47E-04	4.90E-01
2	1.38E-02	3.82E-04	2.91E-01	7.43E-03	3.14E-03	1.75E-02	1.87E-02	3.71E-03	8.90E-02	1.76E-02	1.74E-03	1.55E-01	1.14E-02	1.69E-05	8.87E-01
5	6.58E-03	1.13E-05	7.60E-01	2.33E-03	8.11E-04	6.65E-03	8.77E-03	6.51E-04	1.07E-01	9.49E-03	2.93E-04	2.38E-01	5.74E-03	3.24E-08	9.99E-01
10	3.52E-03	3.84E-07	9.66E-01	8.08E-04	2.05E-04	3.18E-03	4.49E-03	1.02E-04	1.66E-01	5.60E-03	4.42E-05	4.18E-01	3.20E-03	7.03E-11	1.00E+00
20	1.78E-03	6.13E-09	9.98E-01	2.33E-04	3.57E-05	1.52E-03	2.10E-03	9.95E-06	3.07E-01	3.11E-03	4.25E-06	6.96E-01	1.67E-03	4.23E-14	1.00E+00
35	9.75E-04	1.16E-10	1.00E+00	7.31E-05	6.38E-06	8.36E-04	1.05E-03	1.03E-06	5.16E-01	1.85E-03	4.53E-07	8.83E-01	9.32E-04	3.72E-17	1.00E+00
50	6.49E-04	6.72E-12	1.00E+00	3.21E-05	1.81E-06	5.70E-04	6.46E-04	1.99E-07	6.78E-01	1.29E-03	9.14E-08	9.48E-01	6.27E-04	2.43E-19	1.00E+00
100	2.79E-04	2.12E-14	1.00E+00	5.28E-06	1.03E-07	2.72E-04	2.29E-04	4.88E-09	9.15E-01	6.09E-04	2.66E-09	9.93E-01	2.72E-04	3.60E-24	1.00E+00
200	1.10E-04	1.28E-17	1.00E+00	6.34E-07	3.08E-09	1.30E-04	7.01E-05	5.59E-11	9.89E-01	2.63E-04	4.12E-11	9.99E-01	1.07E-04	7.18E-30	1.00E+00
500	2.82E-05	8.08E-23	1.00E+00	2.15E-08	9.05E-12	5.13E-05	1.13E-05	3.71E-14	1.00E+00	7.51E-05	5.50E-14	1.00E+00	2.65E-05	5.73E-39	1.00E+00
600	2.11E-05	5.33E-24	1.00E+00	1.00E-08	2.34E-12	4.30E-05	7.57E-06	6.94E-15	1.00E+00	5.72E-05	1.24E-14	1.00E+00	1.96E-05	5.12E-41	1.00E+00
1000	8.98E-06	4.40E-27	1.00E+00	9.89E-10	3.62E-14	2.70E-05	2.27E-06	4.08E-17	1.00E+00	2.56E-05	1.38E-16	1.00E+00	8.01E-06	3.13E-47	1.00E+00
2000	2.56E-06	1.27E-32	1.00E+00	2.65E-11	4.50E-17	1.57E-05	3.62E-07	1.20E-20	1.00E+00	7.75E-06	1.31E-19	1.00E+00	2.11E-06	6.97E-57	1.00E+00
5000	4.00E-07	1.02E-40	1.00E+00	8.29E-14	7.14E-22	9.62E-06	2.14E-08	2.26E-26	1.00E+00	1.29E-06	2.29E-24	1.00E+00	2.88E-07	3.94E-72	1.00E+00
10000	8.30E-08	1.08E-48	1.00E+00	4.32E-16	2.15E-26	8.68E-06	1.77E-09	1.17E-31	1.00E+00	2.78E-07	1.25E-28	1.00E+00	5.21E-08	7.37E-86	1.00E+00

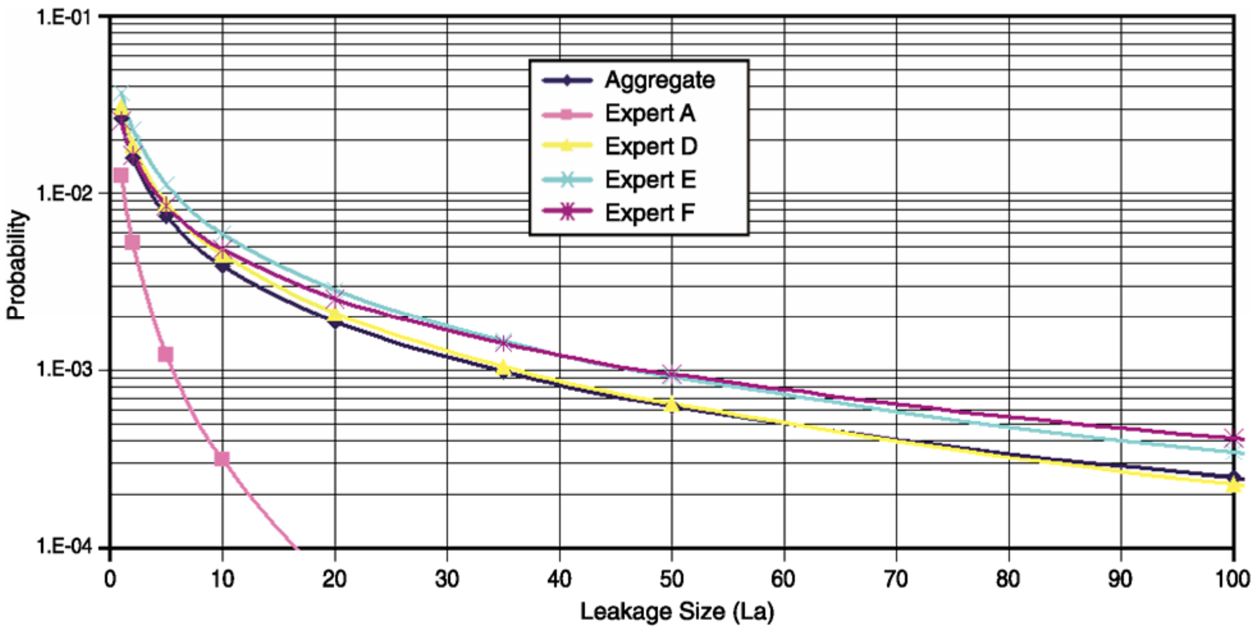

Figure F-1
Large Containment – All Failure Modes

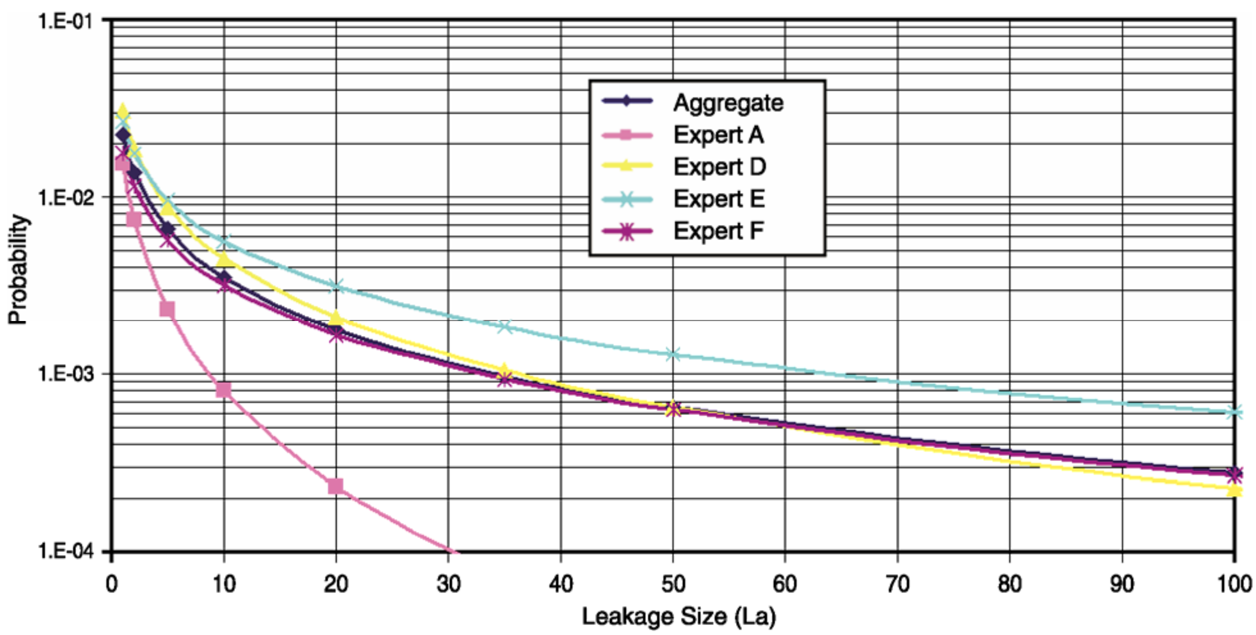

Figure F-2
Small Containment – All Failure Modes

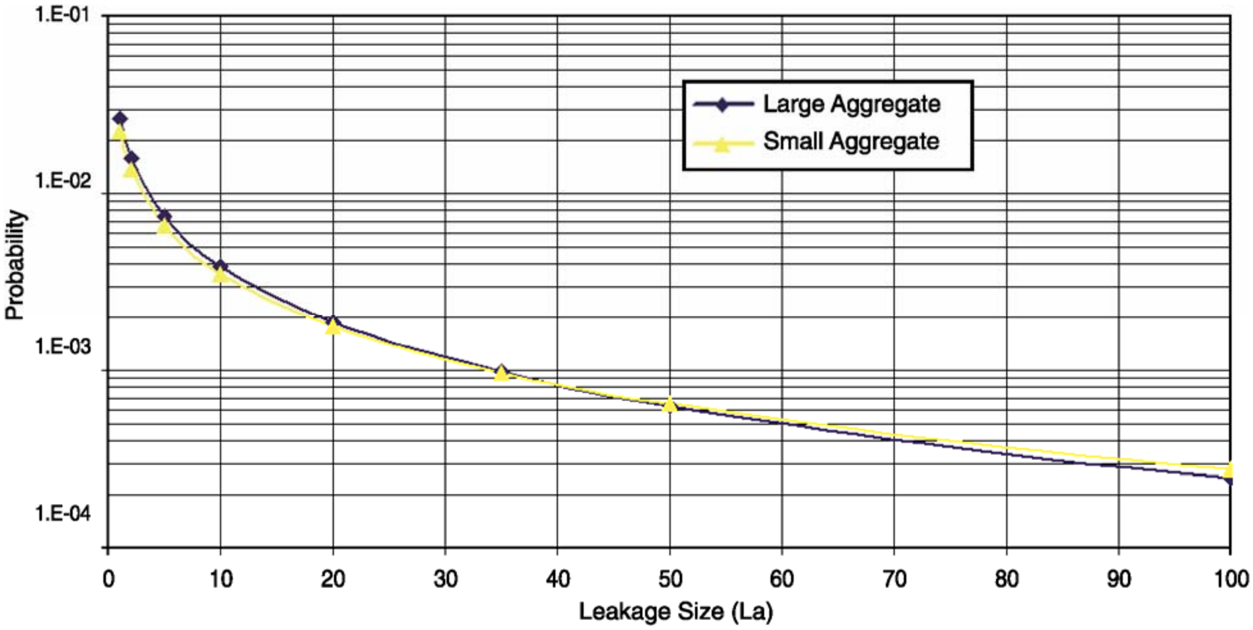

Figure F-3
Comparison of Small and Large Containment – Failure Probability

G
SUMMARY OF ILRT SUBMITTALS

This appendix provides a summary of the one-time ILRT extension submittals that have been made to the NRC. A description of Table G-1 is as follows:

- Column 1 of the table provides an item number.
- Column 2 provides the dates of the various utility submittals to the NRC, including requests for additional information (RAIs) or other correspondence that relates to the submittal.
- Column 3 provides the date of the approval of the submittal. This column is completed if the submittal has been approved.
- Column 4 provides the plant name.
- Column 5 provides the reactor type. Reactor types are Westinghouse PWR (PWR West), Combustion Engineering PWR (PWR CE), General Electric BWR (BWR-X, where X is the model), and Babcock & Wilcox PWR (PWR B&W).
- Column 6 provides a description of containment type.
- Column 7 provides the delta LERF. Notes are provided where significant sensitivity studies are presented. If the total LERF was provided in the submittal, it is also provided in this column.
- Column 8 provides the delta conditional containment failure probability (CCFP) if provided in the submittal. Early one-time extensions did not always require delta CCFP.
- Column 9 provides the population dose. The population dose is expressed in either a person-rem per year increase or as a percent increase of the total population dose.

Notes on specific submittals are provided if warranted by additional information contained in the submittal. These notes are presented in Table G-2.

Table G-1
Summary of One-Time ILRT Extension to 15 Years

No.	Date Submitted	Date Approved	Plant Name	Reactor Type	Containment Type	Delta LERF (per year)	Delta CCFP	Population Dose
1.	09/29/05		Seabrook Extend 15 years to 15 years, 6 months	PWR West.	Large reinforced concrete primary containment structure			
2.	04/06/05		Shearon Harris	PWR West.	Reinforced concrete cylinder with steel liner	7.68E-8	0.312%	0.0154 person-rem per year
3.	03/31/05		St. Lucie 2 From 15 to 15.5 years	PWR CE	Large, dry (ambient) steel primary containment structure; steel containment shell structure with concrete enclosure building	9.5E-8 for 16-year interval	0.49%	0.03 person-rem per year[a]
4.	3/10/05		Ginna Estimated approval of 10/31/2005	PWR West.	Reinforced concrete cylinder w/ steel liner, post-tensioned in three directions	2.44E-8 (Total LERF is 6.44E-6)	0.49%	0.46%
5.	03/08/05		River Bend 1 Extend from 15 to 15 and 4 months.	BWR-6	Mark III			
6.	09/06/00 01/18/01 04/02/01	04/17/01	Indian Point 3	PWR West.	Steel-lined reinforced concrete vertical cylinder w/ flat base mat and hemispherical dome enclosing the entire reactor and RCS	1.1E-7	0.4%	0.43%
7.	03/07/01 04/25/01 06/20/01 07/16/01	08/30/01	Crystal River	PWR B&W	Reinforced concrete cylinder with steel liner	4.4E-8	0.0031%	0.14%

Table G-1 (Continued)
Summary of One-Time ILRT Extension to 15 Years

No.	Date Submitted	Date Approved	Plant Name	Reactor Type	Containment Type	Delta LERF (per year)	Delta CCFP	Population Dose
8.	05/30/01 (two letters) 07/24/01 08/13/01	10/04/01	Peach Bottom Unit 3	BWR 4	Mark I	1.4E-8	0.0031	0.12%
9.	10/17/01	01/29/02	Turkey Point Units 3 & 4	PWR West.	Large, shallow-dome, prestressed concrete containment structure	3.0E-8	0.0029%	0.05%
10.	07/23/01 09/21/01 11/08/01	02/14/02	Waterford 3	PWR CE	Large, dry (ambient) steel primary containment structure	8.0E-8	0.0024%	0.14%
11.	08/31/01 01/24/2002	02/20/02	Hatch 1	BWR GE	Mark I	4.0E-8	0.0032%	0.19%
12.	03/05/01 09/04/01	02/28/02	Oconee 3 extending the Type A test interval to 151 months	B&W PWR	Reinforced concrete cylinder with steel liner	8.2E-8	0.0031%	0.11%
13.	11/26/01 01/31/02 02/05/02 02/11/02	03/06/02	Brunswick 1 (for 3 years, 2 months)	BWR 4	Mark I	1.5E-7 Visual inspections credit for large flaw detection— LERF decreases from 1.5E-7 to 7.5E-8	0.0031%	0.002%
14.	07/30/01 09/07/01 10/16/01 12/05/01 01/18/02	03/08/02	Susque-hanna Units 1 & 2	BWR GE	Mark II steel-lined, reinforced-concrete drywell & pressure suppression chamber (PSC) and an integrally-reinforced, concrete slab separating the drywell from the PSC	1E-9	0.0063%	0.05%

Table G-1 (Continued)
Summary of One-Time ILRT Extension to 15 Years

No.	Date Submitted	Date Approved	Plant Name	Reactor Type	Containment Type	Delta LERF (per year)	Delta CCFP	Population Dose
15.	08/02/01 11/02/01 12/04/01 12/19/01 01/07/02	04/11/02	Seabrook 1	PWR West.	Large reinforced concrete primary containment structure	1.5E-7 both internal and external events (Total LERF is 1.2E-6)	0.0031%	0.03%
16.	11/16/01	04/22/02	Diablo Canyon 1 & 2	PWR West.	Large, dry reinforced concrete primary containment structure	3E-8	0.003%	0.17%
17.	01/31/02 03/27/02	05/01/02	Calvert Cliffs	PWR CE	Post-tensioned, reinforced concrete cylinder and dome	2.9E-7 6.8E-8 (with “effective” inspection assumptions used)	0.0031%	
18.	11/09/01 03/13/02 04/11/02	05/07/02	Sequoyah 2 (11.5-year extension only)	PWR West.	Freestanding steel vessel with an ice condenser, separate secondary containment—reinforced-concrete, shield building	2.0 E-7 Licensee could not quantify external events, so amendment only granted for 1 cycle.	0.0031%	0.37%
19.	01/17/02 03/08/02 03/22/02	06/03/02 (correction to 04/11/02 letter)	Salem 2	PWR West.	Large, reinforced-concrete, vertical right cylinder with a flat base and hemispherical dome	8.6E-8	0.0040%	0.16%
20.	07/13/01 11/30/01 03/13/02 04/03/02 05/30/02 06/13/02	08/05/02	Indian Point 2	PWR West.	Steel-lined reinforced concrete vertical cylinder w/ flat base mat and hemispherical dome enclosing the entire reactor and RCS [b]	9.8E-8 [c]	0.0032	

Table G-1 (Continued)
Summary of One-Time ILRT Extension to 15 Years

No.	Date Submitted	Date Approved	Plant Name	Reactor Type	Containment Type	Delta LERF (per year)	Delta CCFP	Population Dose
21.	12/26/01 02/04/02 06/12/02	08/15/02	Comanche Peak 1 & 2	PWR West.	Large, reinforced concrete vertical right cylinder with a flat base and a hemispherical dome	5.6E-8 5.9E-8 incl. undetectable liner flaw	0.0031%	0.006%
22.	03/26/02 06/19/02 08/08/02	09/16/02	Robinson 2	PWR West.	Dry, post-tension, steel-lined concrete vertical cylinder with a hemispherical dome	1.0E-7 Extension only given for 12.7-year interval—shortened interval so delta LERF would remain in very small region. Note credit given for IWE inspections.	0.006%	0.1%
23.	08/02/01 03/06/02 04/02/02 06/25/02	09/17/02	South Texas 1 & 2	PWR West.	Large, dry concrete shell and steel liner	1.2 E-7 (internal and external) (with corrosion) (Total LERF is 6.1E-6)	0.011%	0.006%
24.	01/31/02 09/09/02	09/24/02	ANO-1	PWR West.	Large, pre-stressed, concrete, vertical right, steel lined cylinder w/ flat base, shallow spherical dome	4.6E-8 (credit for inspection effectiveness, internal only)	0.03%	0.04%
25.	11/26/01 01/31/02 02/05/02 02/11/02 11/08/02	11/21/02	Brunswick 2 (for 2 years, 2 months)	BWR 4	Mark I	1.5E-7 [d]	0.3%	0.002%

Table G-1 (Continued)
Summary of One-Time ILRT Extension to 15 Years

No.	Date Submitted	Date Approved	Plant Name	Reactor Type	Containment Type	Delta LERF (per year)	Delta CCFP	Population Dose
26.	10/15/01 11/08/01 06/28/02 07/25/02.	12/16/02	Surry 1	PWR West.	Subatmospheric containment. The containment consists of a steel-lined reinforced concrete cylinder with hemispherical dome and a reinforced concrete basemat.	1.2E-7 Assuming the visual inspections are capable of detecting large flaws in the visible regions of the containment, then the increase in LERF would go from 1.2E-7 to 1.8E-8.	0.0031%	0.004%
27.	12/07/01 06/28/02 07/25/02	12/31/02	North Anna 1	PWR West.	Large reinforced concrete primary containment structure	1.1E-7 Assuming visual inspections are capable of detecting large flaws in visible regions the containment, increase in LERF drops from 1.1 E-7 to 1.7E-8. When corrosion of containment liner is considered, increase in LERF is estimated to be 2.1E-8.	0.3%	0.003%
28.	04/11/02 11/11/02	02/25/03	DC Cook Units 1 & 2	PWR West.	Ice condenser; reinforced, concrete vessel with a steel liner	1.23E-7 1.32E-7 when possible corrosion of containment surfaces is considered.	0.3%	0.03%
29.	10/31/02 12/02/02 01/24/03	03/05/03	Beaver Valley 1 & 2	PWR West.	Reinforced concrete, steel-lined vessels with a flat base, cylindrical walls, and a hemispherical dome	2.1E-7 (Unit 1) 3.8E-8 (Unit 2) (without corrosion) 2.2E-7 (Unit 1) 3.9E-8 (Unit 2) (with corrosion)	0.3% (both units)	0.04% (Unit 1) 0.02% (Unit 2)

Table G-1 (Continued)
Summary of One-Time ILRT Extension to 15 Years

No.	Date Submitted	Date Approved	Plant Name	Reactor Type	Containment Type	Delta LERF (per year)	Delta CCFP	Population Dose
30.	05/14/02 12/20/02	03/05/03	River Bend	BWR 6	Mark III	3.0E-8	0.3%	0.3%
31.	05/29/02 09/25/02 11/12/02 01/08/03 01/29/03	03/12/03	Catawba 1 & 2	PWR West.	Ice condenser; a freestanding cylindrical steel structure enclosed by a separate reinforced-concrete reactor building	5.1E-7 (Total LERF is 6.55E-6)	0.7%	0.04%
32.	05/29/02 09/25/02 11/12/02 01/08/03 01/29/03	03/12/03	McGuire 1 & 2	PWR West.	Ice-condenser primary containment structure; free-standing cylindrical steel structure enclosed by a separate reinforced-concrete reactor building	4.4E-7 (Total LERF is 4.57E-6)	0.8%	0.07 person-rem/year
33.	05/23/02 12/20/02 02/27/03	03/27/03	Fermi	BWR-4	Mark I	1.7E-8	0.5%	0.1%
34.	03/29/02 01/24/03	03/31/03	Duane Arnold	BWR-4	Mark I	3.7E-8	0.4%	0.24%
35.	04/22/02 10/25/02 01/23/03 02/12/03	03/31/03	Monticello	BWR-3	Mark I	1.7E-7	1.1%	0.2 person-rem per year
36.	04/04/02 01/09/03	03/31/03	Farley 1 & 2	PWR West.	Large, prestressed, reinforced-concrete, primary containment structure	2.0E-7 (Unit 1) 4.1E-7 (Unit 2) (Total LERF is 1E-6)	0.5% Unit 1 0.7% Unit 2	< 0.01 person-rem/year

Table G-1 (Continued)
Summary of One-Time ILRT Extension to 15 Years

No.	Date Submitted	Date Approved	Plant Name	Reactor Type	Containment Type	Delta LERF (per year)	Delta CCFP	Population Dose
37.	03/14/02 01/20/03	04/08/03	Perry	BWR 6	Mark III	6.4E-8 When possible corrosion of containment surfaces is considered increase in LERF is estimated to be 6.9E-8	1%	0.04%
38.	08/15/02 12/13/02	04/10/03	St. Lucie 1 & 2	PWR CE	Large, dry (ambient) steel primary containment structure; steel containment shell structure with concrete enclosure building,	9.4E-8 (Unit 1) 7.7E-8 (Unit 2) (without corrosion) 1.1E-7 (Unit 1) 8.3E-8 (Unit 2) (considering corrosion)	0.3 %	0.18% (Unit 1) 0.11% (Unit 2)
39.	10/09/02 11/22/02 12/06/02	04/16/03	Hope Creek	BWR-4	Mark I	5.2E-8	0.6%	0.2%
40.	10/04/02 02/19/03 05/19/03	05/29/03	Sequoyah 1 & 2	PWR West.	Freestanding steel vessel with an ice condenser, separate secondary containment—reinforced-concrete, shield building	6.5E-8 (without corrosion) 1E-7 (with corrosion)	0.3%	0.37%
41.	10/04/02	06/02/03	Vermont Yankee	BWR/4	Mark I	No risk analysis information used in SER.		
42.	09/30/02 03/19/03	08/14/03	TMI 1	PWR B&W	Post-tensioned reinforced concrete cylinder and spherical dome, inner surfaces completely lined with welded steel plate.	1.7E-7 (internal) 6.3E-7 (int. and ext.) (Total LERF is 5.1E-6)	1.0%	0.2 person-rem per year

Table G-1 (Continued)
Summary of One-Time ILRT Extension to 15 Years

No.	Date Submitted	Date Approved	Plant Name	Reactor Type	Containment Type	Delta LERF (per year)	Delta CCFP	Population Dose
43.	10/08/02 04/11/03 05/21/03	08/15/03	Ft. Calhoun	PWR CE	Steel-lined pre-stressed concrete containment	4.2E-8 (internal events PRA that includes major risk contributors from seismic events)	<0.1%	0.16%
44.	10/24/02 06/20/03	11/19/03	LaSalle 1 & 2	BWR 5	Mark II	3.0E-8 LERF increase associated with corrosion events is estimated to be less than 1E-8 per year.	0.5%	0.08 person-rem per year
45.	01/29/03 09/15/03	01/08/04	Clinton	BWR 6	Mark III	3.0E-7 (internal events)[e] (Total LERF is 8E-7)	1.1%	< 0.1 person-rem per year
46.	02/26/03 07/25/03	01/15/04	Vogtle 1 & 2	PWR West.	Large, prestressed, reinforced-concrete, primary containment structure.	1.7E-7 (internal) (Total LERF is 6E-7)	1%	< 0.1 person-rem per year
47.	05/12/03 10/29/03	01/28/04	Grand Gulf	BWR-6	Mark III	7.7E-8 [f]	1.0%	< 0.1 person-rem per year
48.	06/11/03 08/20/03 10/13/03	02/11/04	Robinson	PWR West.	Dry, post-tension, steel-lined concrete vertical cylinder with a hemispherical dome	1.3E-7 (internal)	0.6%	0.01 person-rem
49.	02/27/03 04/11/03 08/05/03	03/08/04	Quad Cities Units 1 & 2	BWR-3	Mark I	1.3E-8 (Internal events) increase in LERF associated with corrosion events is estimated to be < 1E-8	0.6%	<0.01 person-rem per year.

Table G-1 (Continued)
Summary of One-Time ILRT Extension to 15 Years

No.	Date Submitted	Date Approved	Plant Name	Reactor Type	Containment Type	Delta LERF (per year)	Delta CCFP	Population Dose
50.	06/20/03 12/12/03	04/06/04	Kewaunee	PWR West.	Large, dry (ambient) steel primary containment structure; a steel containment shell structure w/ a concrete enclosure building	7.6E-7 (internal) (Total LERF is 8E-6)	0.4%	0.03 person-rem per year
51.	07/28/03 05/20/04	09/28/04	Fitzpatrick	BWR 4	Mark I	2.6E-8 Associated with corrosion events is 1E-8	1.1%	0.01 person rem per year
52.	01/15/04 06/22/04	10/13/04	Dresden Units 2 & 3	BWR-3 GE	Mark I	1.8 E-8 (Internal events) increase in LERF associated with corrosion events is estimated to be less than 1E-8 per year.	1.0%	< 0.01 person-rem per year [g]
53.	04/26/04 08/17/04 09/07/04	02/01/05	Hatch 2	BWR-4 GE	Mark I	1.1 E-7 Total = 3 E-6 (internal and external)	1%	0.03 person-rem per year.
54.	07/08/04 11/24/04	03/09/05	Browns Ferry 2 & 3	BWR-4 GE	Mark I	3.4E-8 (internal) 6.2E-8 (internal and external)	1%	< 0.01 person-rem per year
55.	04/14/04 11/10/04	03/30/05	Pilgrim	BWR 3 GE	Mark I	4.7E-9 (internal) 2.6E-7 (int. and ext.) (Total LERF is 7E-6)	<0.1%	< 0.01 person-rem/year
56.	07/06/04 09/21/04 12/23/04	04/06/05	Millstone 2	PWR CE	Steel-lined concrete containment	7.8E-7 (Internal) (Total LERF is 3E-6)	1%	< 0.01 person-rem per year
57.	08/05/04 01/17/2005	04/12/05	Columbia	BWR-5 GE	Mark II	4.7E-9 (internal) 2.6E-7 (internal and external)	< 1%	< 0.01 person-rem per year

Table G-1 (Continued)
Summary of One-Time ILRT Extension to 15 Years

No.	Date Submitted	Date Approved	Plant Name	Reactor Type	Containment Type	Delta LERF (per year)	Delta CCFP	Population Dose
58.	06/30/2004 12/2/04 05/27/05 07/18/05	08/24/05	SONGS 2 & 3	PWR CE	Reinforced concrete cylinder w/ steel liner, post-tensioned in 3 directions.	2.0E-7 (internal only) 3.8E-7 (internal and external) (Total LERF 4E-7)	1 %	< 0.01 person-rem per year
59.	10/5/04 04/22/05	08/31/05	Vermont Yankee	BWR/4	Mark I	6.9E-9 (internal) 8.8E-8 (int and ext)	0.1%	0.01 person-rem per year

Table G-2
ILRT Summary Notes

Note	Description
a	This was an extension of their 15-year interval to 15.5 years. Not approved to date by staff.
b	IP2 is one of the very few U.S. plants to have a system that pressurizes the containment weld channels and certain containment penetrations during normal plant operation.
c	Sensitivity case done: incorporated potential degradation in uninspectable side of liner into the risk assessment, the 12% reduction in the containment strength is acceptable. Resultant delta LERF was 1.0E-7 per year.
d	Visual inspections credit for large flaw detection—LERF decreases from 1.5E-7 to 7.5E-8. With corrosion LERF is 9.2E-8/yr.
e	If probability of drywell failure categories is based on Chi-square upper bound value for all Mark III DBLRT tests, the increase in LERF is estimated to be 4.5E-7 per year. Increase in LERF associated with corrosion events is estimated to be approximately 2E-8 per year.
f	3.8E-7 (based on "as-found" DWBT results for all Mark III containments) (credit for drywell leakage mitigation by containment sprays) 2.4E-8 (credit for drywell leakage mitigation by containment sprays or RCS depressurization)
g	Increase from including bellows degradation is estimated about 0.02 person-rem per year based on TS leakage of 0.5% per day and 0.1 person-rem per year based on a leakage of 3.0% per day used in the (pending) AST application for Dresden.

H
RISK IMPACT ASSESSMENT TEMPLATE

The following report appendix contains a template for the performance of the Risk Impact Assessment of Extended Integrated Leak Rate Testing Intervals. The main purpose of the template is to illustrate the types of information that should be included in a plant-specific confirmation of risk impact associated with the extension of ILRT intervals. The template is one suggested approach to performing the assessment. Other approaches are not precluded. In applying the template the analyst should ensure that all relevant information is appropriately documented. In addition, the final assessment should comply with appropriate plant specific procedures for the documentation and control of similar types of assessments.

CONTENTS

1.0 PURPOSE OF ANALYSIS

1.0 PURPOSE

The purpose of this analysis is to provide a risk assessment of extending the currently allowed containment Type A integrated leak rate test (ILRT) to a permanent fifteen years. The extension would allow for substantial cost savings as the ILRT could be deferred for additional scheduled refueling outages for the ______________________. The risk assessment follows the guidelines from NEI 94-01, Revision 2 [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" from November 2001 [3], the NRC regulatory guidance on the use of Probabilistic Risk Assessment (PRA) as stated in Regulatory Guide 1.200 as applied to ILRT interval extensions, and risk insights in support of a request for a plant's licensing basis as outlined in Regulatory Guide (RG) 1.174 [4], 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 [5], and the methodology used in EPRI 1009325, Revision 2 [26].

1.1 BACKGROUND

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

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

The NRC report on performance-based leak testing, NUREG-1493, analyzed the effects of containment leakage on the health and safety of the public and the benefits realized from the containment leak rate testing. In that analysis, it was determined that for a representative PWR plant (i.e., Surry) that containment isolation failures contribute less than 0.1 percent to the latent risks from reactor accidents. Consequently, it is desirable to show that extending the ILRT interval will not lead to a substantial increase in risk from containment isolation failures for the_________________plant.

The Guidance provided in Appendix H of EPRI Report No. 1009325, Revision 2, *Risk Impact Assessment of Extended Integrated Leak Rate Testing Intervals [26]* for performing risk impact assessments in support of ILRT extensions builds on the EPRI Risk Assessment methodology, EPRI TR-104285. This methodology is followed to determine the appropriate risk information for use in evaluating the impact of the proposed ILRT changes.

It should be noted that containment leak-tight integrity is also verified through periodic inservice inspections conducted in accordance with the requirements of the American Society of Mechanical Engineers (ASME) Boiler and Pressure Vessel Code Section XI. More specifically, Subsection IWE provides the rules and requirements for inservice inspection of Class MC pressure-retaining components and their integral attachments, and of metallic shell and penetration liners of Class CC pressure-retaining components and their integral attachments in light-water cooled plants. Furthermore, NRC regulations 10 CFR 50.55a(b)(2)(ix)(E) require licensees to conduct visual inspections of the accessible areas of the interior of the containment. The associated change to NEI 94-01 will require that visual examinations be conducted during at least three other outages, and in the outage during which the ILRT is being conducted. These requirements will not be changed as a result of the extended ILRT interval. In addition, Appendix J, Type B local leak tests performed to verify the leak-tight integrity of containment penetration bellows, airlocks, seals, and gaskets are also not affected by the change to the Type A test frequency.

1.2 CRITERIA

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

Regarding CCFP, changes of up to 1.1% have been accepted by the NRC for the one-time requests for extension of ILRT intervals. In context, it is noted that a CCFP of 1/10 (10%) has been approved for application to evolutionary light water designs. Given these perspectives, a change in the CCFP of up to 1.5% (percentage point) is assumed to be small.

In addition, the total annual risk (person rem/yr population dose) is examined to demonstrate the relative change in this parameter. While no acceptance guidelines for these additional figures of merit are published, examinations of NUREG-1493 and Safety Evaluation Reports (SER) for one-time interval extension (summarized in Appendix G) indicate a range of incremental increases in population dose that have been accepted by the NRC[11]. The range of incremental population dose increases is from ≤.0.01 to 0.2 person-rem/yr and/or 0.002 to 0.46% of the total accident dose. The total doses for the spectrum of all accidents (NUREG-1493 [2], Figure 7-2) result in health effects that are at least two orders of magnitude less than the NRC Safety Goal Risk. Given these perspectives, a very small population dose is defined as an increase from the baseline interval (3 tests per 10 years) dose of ≤1.0 person-rem per year or 1% of the total baseline dose, whichever is less restrictive for the risk impact assessment of the proposed extended ILRT interval.

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

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

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

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

Moreover, it is noted that the CLIIP notice associated with promulgation of NEI 94-01, Revision 2 will indicate that the CLIIP only applies to those plants that do not credit overpressure for ECCS pump operation and a traditional license amendment require is required in those cases where overpressure is credited.

[11] The methodology used in the one-time ILRT interval extension requests assumed a large leak magnitude (EPRI class 3b) of 35La, whereas the methodology in this document uses 100La. The dose rates are impacted by this change and will be larger than those in previous submittals.

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. The approach is consistent with that presented in Appendix H of EPRI Report No. 1009325, Revision 2, *Risk Impact Assessment of Extended Integrated Leak Rate Testing Intervals [26]* , EPRI TR-104285 [2], NUREG-1493 [6] and the Calvert Cliffs liner corrosion analysis [5]. The analysis uses results from a Level 2 analysis of core damage scenarios from the current _______________ PSA model and subsequent containment response resulting in various fission product release categories (including no or negligible release). This risk assessment is applicable to ____________________________.

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 with the acceptance guidelines of RG 1.174.

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

6. Evaluate the sensitivity of the results to assumptions in the liner corrosion analysis, external events and to the fractional contribution of increased large isolation failures (due to liner breach) to LERF.

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

- Consistent with the other industry containment leak risk assessments, the _____________ assessment uses LERF and delta LERF in accordance with the risk acceptance guidance of RG 1.174. Changes in population dose and conditional containment failure probability are also considered to show that defense-in-depth and the balance of prevention and mitigation is preserved.

- If containment overpressure is credited in the ECCS recirculation analysis and is included in the assessment an additional figure of merit is core damage frequency (CDF) to ensure that the guidelines from RG 1.174 are met.
- This evaluation for __________ uses ground rules and methods to calculate changes in risk metrics that are similar to those used in Appendix H of EPRI Report No. 1009325, Revision 2, Risk Impact Assessment of Extended Integrated Leak Rate Testing Intervals.

3.0 GROUND RULES

The following ground rules are used in the analysis:

- The technical adequacy of the ________________________PRA is consistent with the requirements of Regulatory Guide 1.200 as is relevant to this ILRT interval extension.
- The _______________ Level 1 and Level 2 internal events PSA models provide representative results.
- It is appropriate to use the _______________ internal events PSA model as a gauge to effectively describe the risk change attributable to the ILRT extension. It is reasonable to assume that the impact from the ILRT extension (with respect to percent increases in population dose) will not substantially differ if fire and seismic events were to be included in the calculations.
- Dose results for the containment failures modeled in the PSA can be characterized by information provided in NUREG/CR-4551 [7]. They are estimated by scaling the NUREG/CR-4551 results by population differences for _______________ 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 1La. Class 3 accounts for increased leakage due to Type A inspection failures.
- The representative containment leakage for Class 3a sequences is 10La. based on the previously approved methodology performed for Indian Point Unit 3 [8, 9].
- The representative containment leakage for Class 3b sequences is 100La. based on the guidance provided in EPRI Report No. 1009325, Revision 2 .
- The Class 3b can be very conservatively categorized as LERF based on the previously approved methodology [8, 9].
- The impact on population doses from containment bypass scenarios is not altered by the proposed ILRT extension, but is accounted for in the EPRI methodology as a separate entry for comparison purposes. Since the containment bypass contribution to population dose is fixed, no changes 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.
- Where possible, the analysis should include a quantitative assessment of the contribution of external events (e.g., fire and seismic) in the risk impact assessment for extended ILRT intervals. For example, where a licensee possesses a quantitative fire analysis and that analysis is of sufficient quality and detail to assess the impact, the methods used to obtain the impact from internal events should be applied for the external event. If the external event

analysis is not of sufficient quality or detail to directly apply the methodology provided in this document, the quality or detail will be increased or a suitable estimate of the risk impact from the external events should be performed. This assessment can be taken from existing, previously submitted and approved analyses or other alternate method of assessing an order of magnitude estimate for contribution of the external event to the impact of the changed interval.

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 [14]
6. NUREG-1493 [6]
7. EPRI TR-104285 [2]
8. NUREG-1150 [15] and NUREG/CR-4551 [7]
9. NEI Interim Guidance [3][20]
10. Calvert Cliffs liner corrosion analysis [5]
11. EPRI Report No. 1009325, Revision 2, Appendix H

The first study is applicable because it provides one basis for the threshold that could be used in the Level 2 PSA for the size of containment leakage that is considered significant and is to be included in the model. The second study is applicable because it provides a basis of the probability for significant pre-existing containment leakage at the time of a core damage accident. The third study is applicable because it is a subsequent study to NUREG/CR-4220 that undertook a more extensive evaluation of the same database. The fourth study provides an assessment of the impact of different containment leakage rates on plant risk. The fifth study provides an assessment of the impact on shutdown risk from ILRT test interval extension. The sixth study is the NRC's cost-benefit analysis of various alternative approaches regarding extending the test intervals and increasing the allowable leakage rates for containment integrated and local leak rate tests. The seventh study is an EPRI study of the impact of extending ILRT and 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 ___________. The ninth study includes the NEI recommended methodology (promulgated in two letters) for evaluating the risk associated with obtaining a one-time extension of the ILRT interval. The tenth study addresses the impact of age-related degradation of the containment liners on ILRT evaluations. Finally, the eleventh study builds on the previous work and includes a recommended methodology and template for evaluating the risk associated with a permanent 15-year extension of the ILRT interval.

NUREG/CR-3539 [11]

Oak Ridge National Laboratory documented a study of the impact of containment leak rates on public risk in NUREG/CR-3539. This study uses information from WASH-1400 [16] as the basis for its risk sensitivity calculations. ORNL concluded that the impact of leakage rates on LWR accident risks is relatively small.

NUREG/CR-4220 [12]

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

NUREG-1273 [12]

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

NUREG/CR-4330 [13]

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

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

EPRI TR-105189 [14]

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

NUREG-1493 [6]

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

Reduction in ILRT frequency from 3 per 10 years to 1 per 20 years results in an "imperceptible" increase in risk

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

EPRI TR-104285 [2]

Extending the risk assessment impact beyond shutdown (the earlier EPRI TR-105189 study), the EPRI TR-104285 study is a quantitative evaluation of the impact of extending ILRT and LLRT test intervals on at-power public risk. This study combined IPE Level 2 models with NUREG-1150 Level 3 population dose models to perform the analysis. The study also used the approach of NUREG-1493 in calculating the increase in pre-existing leakage probability due to extending the ILRT and LLRT test intervals.

EPRI TR-104285 uses a simplified Containment Event Tree to subdivide representative core damage frequencies into eight classes of containment response to a core damage accident:

1. Containment intact and isolated
2. Containment isolation failures dependent upon the core damage accident
3. Type A (ILRT) related containment isolation failures
4. Type B (LLRT) related containment isolation failures
5. Type C (LLRT) related containment isolation failures
6. Other penetration related containment isolation failures
7. Containment failures due to core damage accident phenomena
8. Containment bypass

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

"… the proposed CLRT [containment leak rate tests] frequency changes would have a minimal safety impact. The change in risk determined by the analyses is small in both absolute and relative terms. For example, for the PWR analyzed, the change is about 0.04 person-rem per year …"

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

NUREG-1150 and the technical basis, NUREG/CR-4551, provide an ex-plant consequence analysis for a spectrum of accidents including a severe accident with the containment remaining intact (i.e., Tech Spec leakage). This ex-plant consequence analysis is calculated for the 50-mile radial area surrounding Surry. The ex-plant calculation can be delineated to total person-rem for each identified Accident Progression Bin (APB) from NUREG/CR-4551. With the ___________ Level 2 model end-states assigned to one of the NUREG/CR-4551 APBs, it is considered adequate to represent _____________. (The meteorology and site differences other than population are assumed not to play a significant role in this evaluation.)

NEI Interim Guidance for Performing Risk Impact Assessments In Support of One-Time Extensions for Containment Integrated Leakage Rate Test Surveillance Intervals [3][20]

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

Calvert Cliffs Response to Request for Additional Information Concerning the License Amendment for a One-Time Integrated Leakage Rate Test Extension [5]

This submittal to the NRC describes a method for determining the change in likelihood, due to extending the ILRT, of detecting liner corrosion, and the corresponding change in risk. The methodology was developed for Calvert Cliffs in response to a request for additional information regarding how the potential leakage due to age-related degradation mechanisms 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. Licensees may consider approved LARs for one-time extensions involving containment types similar to their facility. The ______________ assessment has addressed the plant-specific differences from the Calvert Cliffs design, and how the Calvert Cliffs methodology was adapted to address the specific design features. In the case where no similar analyses has been performed the licensee will use judgment based the available analyses and plant specific features to perform the analysis.

EPRI Report No. 1009325, Revision 2-A, *Risk Impact Assessment of Extended Integrated Leak Rate Testing Intervals* [26]

This report provides a generally applicable assessment of the risk involved in extension of ILRT test intervals to permanent 15-year intervals. Appendix H of this document provides guidance for performing plant-specific supplemental risk impact assessments and builds on the previous EPRI risk impact assessment methodology [2] and the NRC performance-based containment leakage test program [6], and considers approaches utilized in various submittals, including Indian Point 3 (and associated NRC SER) and Crystal River.

The approach included in this guidance document is used in the ____________ 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.

4.2 PLANT-SPECIFIC INPUTS

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

- Level 1 Model results [17]
- Level 2 Model results [17]
- Release category definitions used in the Level 2 Model [18]
- Population within a 50-mile radius [19]
- ILRT results to demonstrate adequacy of the administrative and hardware issues [19] (1)
- Containment failure probability data [18]

Level 1 Model

The Level 1 PSA model that is used for ___________ is characteristic of the as-built plant. The current Level 1 model is a linked fault tree model, and was quantified with the total Core Damage Frequency (CDF) = X.XXE-X/yr. This applies to both Unit 1 and Unit 2.

(11) The two most recent Type A tests at ___________ Unit 1 and Unit 2 have been successful, so the current Type A test interval requirement is 10 years.

Level 2 Model

The Level 2 Model that is used for ____________ 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 ____________ results in terms of release category.

Table 4.2-1
___________ Level 2 PSA Model Release Categories and Frequencies [17, 18]

Release Category	Definition	Frequency/yr
	Total Release Category Frequency	
	Core Damage Frequency (including uncategorized releases)	

Population Dose Calculations

The population dose is calculated by using data provided in NUREG/CR-4551 and adjusting the results for ___________. Each of the release categories from Table 4.2-1 was associated with an applicable Collapsed Accident Progression Bin (APB) from NUREG/CR-4551 (see below). The collapsed APBs are characterized by 5 attributes related to the accident progression. Unique combinations of the 5 attributes result in a set of 7 bins that are relevant to the analysis. The definitions of the 7 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 for Surry associated with each APB from NUREG/CR-4551.

Table 4.2-2
Summary Accident Progression Bin (APB) Descriptions [7]

Summary APB Number	Description
1	CD, VB, Early CF, Alpha Mode Core damage occurs followed by a very energetic molten fuel-coolant interaction in the vessel; the vessel fails and generates a missile that fails the containment as well. Includes accidents that have an Alpha mode failure of the vessel and the containment except those follow Event V or an SGTR. It includes Alpha mode failures that follow isolation failures because the Alpha mode containment failure is of rupture size.
2	CD, VB, Early CF, RCS Pressure > 200psia Core Damage occurs followed by vessel breach. Implies Early CF with the RCS above 200 psia when the vessel fails. Early CF means at or before VB, so it includes isolation failures and seismic containment failures at the start of the accident as well as containment failure at VB. It does not include bins in which containment failure at VB follows Event V or an SGTR, or Alpha mode failures.
3	CD, VB, Early CF, RCS Pressure < 200 psia Core damage occurs followed by vessel breach. Implies Early CF with the RCS below psia when the containment fails. It does not include bins in which the containment failure at VB or an SGTR, or Alpha mode failures.
4	CD, VB, Late CF Core Damage occurs followed by vessel breach. Includes accidents in which the containment was not failed or bypassed before the onset of core-concrete interaction (CCI) and in which the vessel failed. The failure mechanisms are hydrogen combustion during CCI, Basemat Melt-Through (BMT) in several days, or eventual overpressure due to the failure to provide containment heat removal in the days following the accident.
5	CD, Bypass Core Damage occurs followed by vessel breach. Includes Event V and SGTRs no matter what happens to the containment after the start of the accident. It also includes SGTRs that do not result in VB.

6	CD, VB, No CF Core Damage occurs followed by vessel breach. Includes accidents not evaluated in one of the previous bins. The vessel's lower head is penetrated by the core, but the containment does not fail and is not bypassed.
7	CD, No VB Core Damage occurs but is arrested in time to prevent vessel breach. Includes accident progressions that avoid vessel failures except those that bypass the containment. Most of the bins placed in this reduce bin have no containment failure as well as no VB. It also includes bins in which the containment is not isolated at the start of the accident and the core is brought to a safe stable state before the vessel fails.

Table 4.2-3
Calculation of Surry Population Dose Risk at 50 Miles [7]

Collapsed Bin #	Fractional APB Contributions to Risk (MFCR) (1)	NUREG/CR-4551 Population Dose Risk at 50 miles (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	0.029	0.158	1.23E-07	1.28E+06
2	0.019	0.106	1.64E-07	6.46E+05
3	0.002	0.013	2.012E-08	6.46E+05 (5)
4	0.216	1.199	2.42E-06	4.95E+05
5	0.732	4.060	5.00E-06	8.12E+05
6	0.001	0.006	1.42E-05	4.23E+02
7	0.002	0.011	1.91E-05	5.76E+02
Totals	1.000	5.55	4.1E-05	

(1) Mean Fractional Contribution to Risk calculated from the average of two samples delineated in Table 5.1-3 of NUREG/CR-4551.

(2) The total population dose risk at 50 miles from internal events in person-rem is provided as the average of two samples 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 2.5-3. 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.

(5) Assumed population dose at 50 miles for Collapsed Bin #3 equal to that of Collapsed Bin #2. Collapsed Bin Frequency #3 was then back calculated using that value. This does not influence the results of this evaluation since Bin #3 does not appear as part of the results for ___________.

Population Estimate Methodology

The person-rem results in Table 4.2-3 can be used as an approximation of the dose for the _______________ if it is corrected for allowable containment leak rate (La), reactor power level and the population density surrounding ____________.

- La adjustment=
  - La of plant (%w/o/day) ÷La of reference plant (applicable only to those APBs affected by normal leakage) La for Surry is 0.1%w/o/day and La for Peach Bottom is 0.5%w/o/day.
- Power level adjustment=
  - Rated power level of plant (MWt) ÷Rated power level of reference plant
  - The rated power level for Surry is 2441MWt, and the rated power level for Peach Bottom is 3293MWt.

Population density adjustment

The total population within a 50-mile radius of ____________ is ____________ [Ref 19]. 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 ____________.

Total ____________ Population50miles = X.XXE+XX

Surry Population from NUREG/CR-4551 = 1.23E+06

Population Dose Factor = X.XXE+XX / 1.23E+06 =_______

The factors developed above are used to adjust the population dose for the surrogate plant (Surry or Peach Bottom) for ____________. For intact containment endstates, the total population dose factor is as follows:

FIntact = FPopulation * FPower Level * FLeakage

FIntact = ___*____*___

FIntact = ____

For EPRI accident classes not dependent on containment leakage, the population dose factor is as follows:

FOthers = FPopulation * FPower Level

FOthers = _____*_____

FOthers = ____

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. The above adjustments provide an approximation for ___________ of the population doses associated with each of the release categories from NUREG/CR-4551.

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 ___________.

Table 4.2-4
Calculation of ___________ Population Dose Risk at 50 Miles

Accident Progression Bin (APB)	NUREG/CR-4551 Population Dose at 50 miles (person-rem)	Bin Multiplier used to obtain ___________ Population Dose	___________ Adjusted Population Dose at 50 miles (person-rem)
1	1.28E+06		
2	6.46E+05		
3	6.46E+05		
4	4.95E+05		
5	8.12E+05		
6	4.23E+02		
7	5.76E+02		

Application of ___________ PSA 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 ___________ PSA 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 match the ___________ PSA Level 2 release categories to the collapsed APBs. The assignments are shown in Table 4.2-5, along with the corresponding EPRI classes (see below).

Table 4.2-5
___________ Level 2 Model Assumptions for Application to the NUREG/CR-4551 Accident Progression Bins and EPRI Accident Classes

___________ Level 2 Release Category	Definition	NUREG/ CR-4551 APB	EPRI Class

Release Category Definitions

Table 4.2-6 defines the accident classes used in the ILRT extension evaluation, which is consistent with the EPRI 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 4.2-6
EPRI Containment Failure Classification [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.

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 4.2-6, is divided into two sub-classes, Class 3a and Class 3b, representing small and large leakage failures, respectively.

The probability of the EPRI Class 3a and 3b failures is determined consistent with the EPRI Guidance [__]. For Class 3a, the probability is based on the maximum likelihood estimate of failure (arithmetic average) from the available data (i.e., 2 “small” failures in 217 tests leads to 2/217=0.0092). For Class 3b, Jefferys non-informative prior distribution is assumed for no “large” failures in 217 tests (i.e., 0.5/(217+1) = 0.0023).

In a follow on letter [20] 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 ___________, as detailed in Section 5, involves the following:

- 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 ______________ Level 2 PSA analysis.
- A review of Class 1 accident sequences shows that several of these cases involve successful operation of containment sprays. It is assumed that, for calculation of the Class 3b and 3a

frequencies, the fraction of the Class 1 CDF associated with successful operation of containment sprays can also be subtracted. A review of the ___________ accident bins that contribute to Class 1 (___________ release categories __ and __) reveals that sprays are available in x.xx% of the cases. Table 4.3-1 provides a detailed breakdown of the sequences in categories __ and __. Sprays are not credited for any of the other release categories.

Table 4.3-1
___________ Level 2 Sequences Contributing to EPRI Class 1 [17]

Sequence	___________ Release Category	Frequency	Sprays Available?
Total			

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. 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 [9]) because it does not factor in the possibility that the failures could be detected by other tests (e.g., the Type B local leak rate tests that will still occur.) Eliminating this possibility conservatively over-estimates the factor increases attributable to the ILRT extension.

4.4 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 [5]. The Calvert Cliffs analysis was performed for a concrete cylinder and dome and a concrete basemat, each with a steel liner. _______________ has a similar type of containment.

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

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

<u>Assumptions</u>

- Consistent with the Calvert Cliffs analysis, a half failure is assumed for basemat concealed liner corrosion due to the lack of identified failures. (See Table 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 this _______________ 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.

- Consistent with the Calvert Cliffs analysis, the estimated historical flaw probability is also limited to 5.5 years to reflect the years since September 1996 when 10 CFR 50.55a started requiring visual inspection. Additional success data was not used to limit the aging impact of this corrosion issue, even though inspections were being performed prior to this date (and have been performed since the time frame of the Calvert Cliffs analysis), 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 cylinder and dome and 0.11% (10% of the cylinder failure probability) for the basemat. These values were determined from an assessment of the probability versus containment pressure, and the selected values are consistent with a pressure that corresponds to the ILRT target pressure of 37 psig. For _______________, the containment failure probabilities are less than these values at 37 psig [18]. Conservative probabilities of 1% for the cylinder and dome and 0.1% for the basemat are used in this analysis, and 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 Cliffs analysis, the likelihood of leakage escape (due to crack formation) in the basemat region is considered to be less likely than the containment cylinder and dome region. (See Table 4.4-1, Step 4.)
- Consistent with the Calvert Cliffs analysis, a 5% visual inspection detection failure likelihood given the flaw is visible and a total detection failure likelihood of 10% is used. To date, all liner corrosion events have been detected through visual inspection. (See Table 4.4-1, Step 5.) Sensitivity studies are included that evaluate total detection failure likelihood of 5% and 15%, respectively.
- Consistent with the Calvert Cliffs analysis, all non-detectable containment failures are assumed to result in early releases. This approach avoids a detailed analysis of containment failure timing and operator recovery actions.

Analysis

Table 4.4-1
Steel Liner Corrosion Base Case

<table>
<tr><th>Step</th><th>DESCRIPTION</th><th colspan="2">Containment Cylinder and Dome</th><th colspan="2">Containment Basemat</th></tr>
<tr><td>1</td><td>Historical Steel Liner Flaw Likelihood</td><td colspan="2">Events: 2</td><td colspan="2">Events: 0
(assume half a failure)</td></tr>
<tr><td></td><td>Failure Data: Containment location specific (consistent with Calvert Cliffs analysis).</td><td colspan="2">2/(70 * 5.5) = 5.2E-3</td><td colspan="2">0.5/(70 * 5.5) = 1.3E-3</td></tr>
<tr><td rowspan="2">2</td><td rowspan="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).</td><td>Year

1
avg 5-10
15</td><td>Failure Rate
2.1E-3
5.2E-3
1.4E-2</td><td>Year

1
avg 5-10
15</td><td>Failure Rate
5.0E-4
1.3E-3
3.5E-3</td></tr>
<tr><td colspan="2">15 year average = 6.27E-3</td><td colspan="2">15 year average = 1.57E-3</td></tr>
<tr><td>3</td><td>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 [5]).</td><td colspan="2">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, however, the values are calculated based on the 3, 10, and 15 year intervals consistent with the desired presentation of the results.</td><td colspan="2">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, however, the values are calculated based on the 3, 10, and 15 year intervals consistent with desired presentation of the results.</td></tr>
</table>

Table 4.4-1 (Continued)
Steel Liner Corrosion Base Case

Step	DESCRIPTION	Containment Cylinder and Dome	Containment Basemat
4	Likelihood of Breach in Containment Given Steel Liner Flaw The failure probability of the cylinder and dome is assumed to be 1% (compared to 1.1% in the Calvert Cliffs analysis). The basemat failure probability is assumed to be a factor of ten less, 0.1%, (compared to 0.11% in the Calvert Cliffs analysis).	1%	0.1%
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)	**0.00071% (at 3 years)** 0.71% * 1% * 10% **0.0041% (at 10 years)** 4.1% * 1% * 10% **0.0094% (at 15 years)** 9.4% * 1% * 10%	**0.00018% (at 3 years)** 0.18% * 0.1% * 100% **0.0010% (at 10 years)** 1.0% * 0.1% * 100% **0.0024% (at 15 years)** 2.4% * 0.1% * 100%

The total likelihood of the corrosion-induced, non-detected containment leakage is the sum of Step 6 for the containment cylinder and dome and the containment basemat as summarized below for ___________________.

Total Likelihood Of Non-Detected Containment Leakage Due To Corrosion for___________________:
At 3 years: __________ + ____________ = ____________%
At 10 years: __________ + ____________ = ____________%
At 15 years: __________ + ____________ = ____________%

The above factors are applied to those core damage accidents that are not already independently LERF or that could never result in LERF. For example, the 3-in-10 year case is calculated as follows:

- Per Table ____, the EPRI Class 3b frequency is _______/yr.
- As discussed in Section XX, the ______ CDF associated with accidents that are not independently LERF or could never result in LERF is ______/yr-____/yr=______/yr.
- The increase in the base case Class 3b frequency due to the corrosion-induced concealed flaw issue is calculated as _____*_____=_____/yr, where _______ was previously shown above to be the cumulative likelihood of non-detected containment leakage due to corrosion at 3 years.
- The 3-in-10 year Class 3b frequency including the corrosion-induced concealed flaw issue is the calculated as ______/yr+______/yr=_______/yr.

5.0 RESULTS

The application of the approach based on the guidance contained in EPRI Report No. 1009325, Revision 2-A, Appendix H, EPRI-TR-104285 [2] and previous risk assessment submittals on this subject [5, 8, 21, 22, 23] have led to the following results. The results are displayed according to the eight accident classes defined in the EPRI report. Table 5-1 lists these accident classes.

The analysis performed examined ____________-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 5-1
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 (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 5-1.

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

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

For the assessment of ILRT impacts on the risk profile, the potential for pre-existing leaks is included in the model. (These events are represented by the Class 3 sequences in EPRI TR-104285). The question on containment integrity was modified to include the probability of a liner breach or bellows failure (due to excessive leakage) at the time of core damage. Two failure modes were considered for the Class 3 sequences. These are Class 3a (small breach) and Class 3b (large breach).

The frequencies for the severe accident classes defined in Table 5-1 were developed for ___________ by first determining the frequencies for Classes 1, 2, 7 and 8 using the categorized sequences and the identified correlations shown in Table 4.2-5, scaling these frequencies to account for the uncategorized sequences, determining the frequencies for Classes 3a and 3b, and then determining the remaining frequency for Class 1. 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 total frequency of the categorized sequences is X.XXE-X/yr, and the total CDF is X.XXE-X, so the scale factor is X.XXX. Table 5-2 contains the frequencies from the categorized sequences, and the resulting frequencies due to the scale factor. The results are summarized below and in Table 5-3.

Table 5-2
______________ Categorized Accident Classes and Frequencies

EPRI Class	__________ Release Category	Frequency Based on Categorized Results (per yr)	Adjusted Frequency Using Scale Factor of X.XXX (per yr)
1			
2			
7			
8			
	Total Frequency		

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 is initially determined from the Level 2 Release Categories __ and __ listed in Table 5-2, minus the EPRI Class 3a and 3b frequency, calculated below.

Class 2 Sequences. This group consists of all core damage accident progression bins for which a failure to isolate the containment occurs. The frequency per year for these sequences is obtained from the Release Category __, listed in Table 5-2.

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 (in excess of design allowable but <10La) or large (>100La).

The respective frequencies per year are determined as follows:

$PROB_{class\_3a}$ = probability of small pre-existing containment liner leakage

= 0.0092 [see Section 4.3]

$PROB_{class\_3b}$ = probability of large pre-existing containment liner leakage

= 0.0023 [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), or that would include containment spray operation such that a Large Release would be unlikely (i.e., x.xx% of the ____________ Release Categories __ and __).

CLASS_3A_FREQUENCY = 0.0092 ∗ (CDF-Class 2-Class 8–0.0xxx*Class 1)

= 0.0092*(_______________________________________)=_________/yr

CLASS_3B_FREQUENCY = 0.0023 ∗ (CDF-Class 2-Class 8–0.0xxx*Class 1)

=0.0023 * (_______________________________________) = __________/yr

For this analysis, the associated containment leakage for Class 3A is $10L_a$ and for Class 3B is $100L_{a.}$ These assignments are consistent with the guidance provided in EPRI Report No. 1009325, Revision 2-A.

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 guidance provided in EPRI Report No. 1009325, Revision 2-A. 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 (e.g., overpressure). For this analysis, the frequency is determined from Release Category __ from the ____________ Level 2 results.

Class 8 Sequences. This group consists of all core damage accident progression bins in which containment bypass occurs. For this analysis, the frequency is determined from Release Categories __ and __ from the ____________ Level 2 results.

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 definitions of accident classes defined in EPRI-TR-104285 the NEI Interim Guidance, and guidance provided in EPRI Report No. 1009325, Revision 2-A. . Table 5-3 summarizes these accident frequencies by accident class for ___________.

Table 5-3
Radionuclide Release Frequencies as a Function of Accident Class (__________ Base Case)

Accident Classes (Containment Release Type)	Description	Frequency (per Rx-yr)	
		EPRI Methodology	EPRI Methodology Plus Corrosion
1	No Containment Failure		
2	Large Isolation Failures (Failure to Close)		
3a	Small Isolation Failures (liner breach)		
3b	Large Isolation Failures (liner breach)		
4	Small Isolation Failures (Failure to seal –Type B)		
5	Small Isolation Failures (Failure to seal—Type C)		
6	Other Isolation Failures (e.g., dependent failures)		
7	Failures Induced by Phenomena (Early and Late)		
8	Bypass (Interfacing System LOCA)		
CDF	All CET end states		

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 containment failure classification are as follows:

Class 1 =	**XXX person-rem (at 1.0L_a)**	**=**	**XXX person-rem (1)**	
Class 2 =	X.XXE+XX (2)			
Class 3a	=	XXX person-rem x 10L_a =	X.XXE+XX person-rem (3)	
Class 3b	=	XXX person-rem x 100L_a	=	X.XXE+XX person-rem (3)
Class 4 =	Not analyzed			
Class 5 =	Not analyzed			
Class 6 =	Not analyzed			
Class 7 =	X.XXE+XX person-rem (4)			
Class 8 =	X.XXE+XX person-rem (5)			

(1) The derivation is described in Section 4.2 for ___________. Class 1 is assigned the dose from the "no containment failure" APBs from NUREG/CR-4551 (i.e., APB #6 and APB #7). The dose is calculated as a weighted average of the dose for these bins using the CDFs for categories __ and __.

(2) The Class 2, containment isolation failures, dose is assigned from APB #2 (Early CF).

(3) The Class 3a and 3b dose are related to the leakage rate as shown. This is consistent with the guidance provided in EPRI Report No. 1009325, Revision 2-A.

(4) The Class 7 dose is assigned from APB #4 (Late CF).

(5) Class 8 sequences involve containment bypass failures; as a result, the person-rem dose is not based on normal containment leakage. The releases for this class are assigned from APB #5 (Bypass).

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] as modified by EPRI Report No. 1009325, Revision 2-A are provided in Table 5-4.

Table 5-4
____________ Population Dose Estimates for Population Within 50 Miles

Accident Classes (Containment Release Type)	Description	Person-Rem (50 miles)
1	No Containment Failure	
2	Large Isolation Failures (Failure to Close)	
3a	Small Isolation Failures (liner breach)	
3b	Large Isolation Failures (liner breach)	
4	Small Isolation Failures (Failure to seal –Type B)	
5	Small Isolation Failures (Failure to seal—Type C)	
6	Other Isolation Failures (e.g., dependent failures)	
7	Failures Induced by Phenomena (Early and Late)	
8	Bypass (Interfacing System LOCA)	

The above dose estimates, when combined with the results presented in Table 5-3, yield the ____________ baseline mean consequence measures for each accident class. These results are presented in Table 5-5.

Table 5-5
___________ 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)	EPRI Methodology		EPRI 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)						
2	Large Isolation Failures (Failure to Close)						
3a	Small Isolation Failures (liner breach)						
3b	Large Isolation Failures (liner breach)						
4	Small Isolation Failures (Failure to seal –Type B)						
5	Small Isolation Failures (Failure to seal—Type C)						
6	Other Isolation Failures (e.g., dependent failures)						
7	Failures Induced by Phenomena (Early and Late)						
8	Bypass (Interfacing System LOCA)						
CDF	All CET end states						

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.

5.3 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).

Risk Impact Due to 10-year Test Interval

As previously stated, Type A tests impact only Class 3 sequences. For Class 3 sequences, the release magnitude is not impacted by the change in test interval (a small or large breach remains the same, even though the probability of not detecting the breach increases). Thus, only the frequency of Class 3a and 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 5-6.

Risk Impact Due to 15-Year Test Interval

The risk contribution for a 15-year interval is calculated in a manner similar to the 10-year interval. The difference is in the increase in probability of leakage in Classes 3a and 3b. For this case, the value used in the analysis is a factor of 5.0 compared to the 3-year interval value, as described in Section 4.3. The results for this calculation are presented in Table 5-7.

Table 5-6
____________ 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)	EPRI Methodology		EPRI 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)						
2	Large Isolation Failures (Failure to Close)						
3a	Small Isolation Failures (liner breach)						
3b	Large Isolation Failures (liner breach)						
4	Small Isolation Failures (Failure to seal –Type B)						
5	Small Isolation Failures (Failure to seal—Type C)						
6	Other Isolation Failures (e.g., dependent failures)						
7	Failures Induced by Phenomena (Early and Late)						
8	Bypass (Interfacing System LOCA)						
CDF	All CET end states						

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 5-7
____________ 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)	EPRI Methodology		EPRI 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)						
2	Large Isolation Failures (Failure to Close)						
3a	Small Isolation Failures (liner breach)						
3b	Large Isolation Failures (liner breach)						
4	Small Isolation Failures (Failure to seal – Type B)						
5	Small Isolation Failures (Failure to seal—Type C)						
6	Other Isolation Failures (e.g., dependent failures)						
7	Failures Induced by Phenomena (Early and Late)						
8	Bypass (Interfacing System LOCA)						
CDF	All CET end states						

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

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

Regulatory Guide 1.174 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 ______________, 100% of the frequency of Class 3b sequences can be used as a very conservative first-order estimate to approximate the potential increase in LERF from the ILRT interval extension (consistent with the EPRI guidance methodology). Based on a ten-year test interval from Table 5-6, the Class 3b frequency is X.XXE-X/yr; and, based on a fifteen-year test interval from Table 5-7, it is X.XXE-X. 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 X.XXE-X/yr. Similarly, the increase due to increasing the interval from 10 to 15 years is X.XXE-X/yr. As can be seen, even with the conservatisms included in the evaluation (per the EPRI methodology), the estimated change in LERF for is below the threshold criteria for a very small change when comparing the 15 year results to the current 10-year requirement, and just above that criteria when compared to the original 3-year requirement.

5.5 STEP 5 – DETERMINE THE IMPACT ON THE CONDITIONAL CONTAINMENT FAILURE PROBABILITY (CCFP)

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 EPRI Report No. 1009325, Revision 2-A. The NRC has previously accepted similar calculations [9] as the basis for showing that the proposed change is consistent with the defense-in-depth philosophy.

$$CCFP = [1 - (\text{Class 1 frequency} + \text{Class 3a frequency}) / CDF] * 100\%$$

`

$$CCFP_3 = X.XX\%$$

$$CCFP_{10} = X.XX\%$$

$$CCFP_{15} = X.XX\%$$

$$\Delta CCFP = CCFP_{15} – CCFP_3 = X.XX\%$$

$$\Delta CCFP = CCFP_{15} - CCFP_{10} = X.XX\%$$

The change in CCFP of slightly more than ______% 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 ______________ are summarized in Table 5-8.

Table 5-8
___________ 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							
2							
3a							
3b							
7							
8							
Total							
ILRT Dose Rate from 3a and 3b							
Delta Total Dose Rate	From 3 yr	N/A					
	From 10 yr	N/A		N/A			
% change in dose rate from base	From 3 yr	N/A					
	From 10 yr	N/A		N/A			
3b Frequency (LERF)							
Delta LERF	From 3 yr	N/A					
	From 10 yr	N/A		N/A			
CCFP %							
Delta CCFP %	From 3 yr	N/A					
	From 10 yr	N/A		N/A			

6.0 SENSITIVITIES

6.1 SENSITIVITY TO CORROSION IMPACT ASSUMPTIONS

The results in Tables 5-5, 5-6 and 5-7 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 cylinder and dome and the basemat were increased and decreased by an order of magnitude. The total detection failure likelihood was adjusted from 10% to 15% and 5%. The results are presented in Table 6-1. 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 X.XXE-X /yr. The results indicate that even with very conservative assumptions, the conclusions from the base analysis would not change.

Table 6-1
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 (1% Cylinder, 0.1% Basemat)	Base Case 10%		
Doubles every 2 yrs	Base	Base		
Doubles every 10 yrs	Base	Base		
Base	Base	*15%*		
Base	Base	*5%*		
Base	10% Cylinder, 1% Basemat	Base		
Base	0.1% Cylinder, 0.01% Basemat	Base		

Table 6-1 (Continued)
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**
Lower Bound				
Doubles every 10 yrs	0.1% Cylinder, 0.01% Basemat	5% 1%		
Upper Bound				
Doubles every 2 yrs	10% Cylinder, 1% Basemat	15% 100%		

6.2 SENSITIVITY TO CLASS 3B CONTRIBUTION TO LERF

The Class 3b frequency for the base case of a three in ten-year ILRT interval is X.XXE-X/yr [Table 5-5]. Extending the interval to one in ten years results in a frequency of X.XXE-X/yr [Table 5-6]. Extending it to one in fifteen years results in a frequency of X.XXE-X/yr [Table 5-7], which is an increase of X.XXE-X/yr. If 100% of the Class 3b sequences are assumed to have potential releases large enough for LERF, then the increase in LERF due to extending the interval from three in ten to one in fifteen is above the RG 1.174 threshold for very small changes in LERF of 1E-7/yr.

6.3 POTENTIAL IMPACT FROM EXTERNAL EVENTS CONTRIBUTION

Note: Where possible, the analysis should include a quantitative assessment of the contribution of external events (e.g., fire and seismic) in the risk impact assessment for extended ILRT intervals. For example, where a licensee possesses a quantitative fire analysis and that analysis is of sufficient quality and detail to assess the impact, the methods used to obtain the impact from internal events should be applied for the external event. If the external event analysis is not of sufficient quality or detail to directly apply the methodology provided in this document, the quality or detail will be increased or a suitable estimate of the risk impact from the external events should be performed. This assessment can be taken from existing, previously submitted and approved analyses or other alternate method of assessing an order of magnitude estimate for contribution of the external event to the impact of the changed interval.

In the ______________ IPEEE, the dominant risk contributor from external events was found to be from fire events. Other potential contributors such as seismic and high winds were found to be within acceptable limits.

At the time of the IPEEE, the ______________ internal events CDF was X.XXE-XX/reactor-year (single model for both units) and the calculated fire CDF was X.XXE-XX/reactor-year. A fire LERF was not calculated for the IPEEE [24].

At the time of the fire analysis, LOSP was the dominant contributor to core damage in the ______________ PRA. The fire high risk areas involved the main control room, switchgear rooms, and other areas affecting electrical power supply and control (electrical raceways, cable spreading, and electrical penetration rooms) in which a fire could lead to an SBO causing a loss of RCP seal cooling resulting in core uncovery due to a seal LOCA.

Since the IPEEE, the ______________ PRA has been converted from a large event tree model to a linked fault tree model using CAFTA software. Due to the PRA conversion process and four subsequent updates, LOSP is no longer the dominant contributor to internal events CDF, which has been reduced to X.XXE-XX/reactor year. The internal events CDF is now dominated by a complete loss of nuclear service cooling water (NSCW) special initiating event. A complete loss of NSCW causes a loss of all RCP seal cooling resulting in a RCP seal LOCA, leading to core uncovery. Therefore, it is reasonable to assume that the External Events CDF can be approximated as no greater than the current Internal Events CDF for calculating the potential impact of the ILRT extension.

For ______________, the reported total Internal Events LERF as determined from a simplified LERF model is X.XXE-XX/reactor-year [24]. Table 5-2 from this analysis provides an estimated total Internal Events LERF value of X.XXE-X/reactor-year. There are some known conservatisms in the simplified LERF model and truncation value impacts that account for this difference, but the higher value will be used in the discussion below for illustration purposes.

Additionally, the External Events baseline LERF would be expected to be less than the Internal Events baseline LERF because some of the Internal Events baseline LERF comes from events that are not events that are initiated by fires (i.e., ISLOCA and SGTR). However, as shown below, even if it is conservatively assumed that the External Events baseline LERF is equivalent to the Internal Events baseline LERF, the total LERF would still be far below the Regulatory Guide 1.174 criteria of 1.0E-05 following the ILRT extension.

The results from these calculations are shown in Table 6-2.

Table 6-2
_______________ Estimated Total LERF Including External Events Impact

Contributor	EPRI Directly (With 100% of Class 3b to LERF from ILRT)
Internal Events LERF	
External Events LERF	
Internal Events LERF due to ILRT (at 15 years)	
External Events LERF due to ILRT (at 15 years)	
Total:	

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 LERF resulting from a change in the Type A ILRT test interval from three in ten years to one in fifteen years is very conservatively estimated as X.XX/yr using the EPRI guidance as written,. As such, the estimated change in LERF is determined to be "very small" using the acceptance guidelines of Reg. Guide 1.174.

- Regulatory Guide 1.174 [4] also states that when the calculated increase in LERF is in the range of 1.0E-06 per reactor year to 1.0E-07 per reactor year, applications will be considered only if it can be reasonably shown that the total LERF is less than 1.0E-05 per reactor year. An additional assessment of the impact from external events was also made. In this case, the total LERF was conservatively estimated as X.XXE-XX for _______________. This is well below the RG 1.174 acceptance criteria for total LERF of 1.0E-05.

- The change in Type A test frequency to once-per-fifteen-years, measured as an increase to the total integrated plant risk for those accident sequences influenced by Type A testing, is X.XX person-rem/yr. EPRI Report No. 1009325, Revision 2-A states that a very small population dose is defined as an increase of ≤ 1.0 person-rem per year or ≤1 % of the total population dose, whichever is less restrictive for the risk impact assessment of the extended ILRT intervals. Moreover, the risk impact when compared to other severe accident risks is negligible.

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

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

Previous Assessments

The NRC in NUREG-1493 [4] 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 ______ confirm these general findings on a plant specific basis considering the severe accidents evaluated for _______, the _______ containment failure modes, and the local population surrounding the ____________________________.

8.0 TEMPLATE REFERENCES

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[17] (Plant reference Level 1 PRA)

[18] (Plant reference Level 2 PRA)

[19] (Plant Reference, Offsite Population distribution, ILRT results)

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[24] (Plant reference: IPEEE)

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