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Engineering Report Cover Sheet

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Risk Impact Assessment of Extending Containment Type A Test Interval

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TABLE OF CONTENTS

TABLE OF CONTENTS (continued)

LIST OF FIGURES

LIST OF TABLES

LIST OF TABLES

EXECUTIVE SUMMARY

In October 26, 1995, the Nuclear Regulatory Commission (NRC) revised 10 CFR 50, Appendix J. The revisions to Appendix J allow licensees to choose containment leakage testing under Option A "Prescriptive Requirements" or Option B "Performance-Based Requirements," for leakage-rate testing of light-water-cooled containments.

The adoption of the Option B performance-based containment leakage rate-testing program did not alter the basic method by which Appendix J leakage rate testing is performed, but did alter the frequency of measuring primary containment leakage in Type A, B and C tests. Frequency is based upon an evaluation which looks at the "as found" leakage history to determine a frequency for leakage testing which provides assurance that leakage limits will be maintained. The changes to Type A test frequency allowed by Option B do not directly result in an increase in containment leakage, only the interval at which such leakage is measured on an integrated basis.

Under Option B, the Integrated Leak Rate Testing (ILRT) Type A surveillance testing requirements was extended from three-in-ten years to at least once per ten years. The revised Type A test 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 is less than the maximum allowable containment leakage limit of $1.0L_a$.

In accordance with the revised containment leakage-rate testing for Appendix J, the Pilgrim Nuclear Power Station (Pilgrim Station) selected the requirements under Option B as its testing program. Pilgrim Station current ten-year Type A test is due to be performed during refueling outage fifteen (RFO15, scheduled for April/May 2005). However, prior to the performance of that test, the Pilgrim Station seeks a one-time exemption based on the substantial cost savings of removing 2 days of critical path time from RFO 15 and therefore, allows deferral of the associated costs out to RFO 17 in 2009. In addition, this initiative directly supports site goals related to capacity factor and World Association of Nuclear Operators (WANO) performance by shortening planned outage duration for RFO 15

The basis for the Option B 10-year test interval is provided in NEI 94-01, "Industry Guideline for Implementing Performance-Based Option of 10 CFR Part 50, Appendix J". This document is based upon a generic evaluation documented in NUREG-1493, "Performance-Based Containment Leak-Test Program", as the technical basis to support regulatory rulemaking in revising the testing requirements to Appendix J, Option B. NUREG-1493 report examined the impact of containment leakage on public health and safety. NUREG-1493 made the following observations with regard to extending the test frequency:

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

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

NUREG-1493 analyzed both Boiling Water Reactors (Peach Bottom and Grand Gulf) and Pressurized Water Reactors (Surry, Sequoyah, and Zion). For Peach Bottom, (a comparable Boiling Water Reactor plant to Pilgrim Station), it was found that increasing the containment leak rates several orders of magnitude over the design basis (0.5 percent per day to 50 percent per day), results in a negligible increase in total population exposure. Therefore, extending the ILRT interval does not result in any significant increase in risk.

In this report, an evaluation is performed to assess the risk impact of extending the current containment Type A ILRT interval. In performing the risk assessment evaluation, the Pilgrim Station risk assessment was performed following the guidelines of NEI 94-01, "Industry Guideline for Implementing Performance-Based Option of 10 CFR Part 50, Appendix J", the methodology used in EPRI TR-104285, "Risk Assessment of Revised Containment Leak Rate Testing Intervals," and the guidance provided in NRC Regulatory Guide 1.174, "An Approach for Using Probabilistic Risk Assessment In Risk-Informed Decisions On Plant-Specific Changes to the Licensing Basis". The assessment also followed the guidance and additional information distributed by NEI in November 2001 to their Administrative Points of Contact regarding risk assessment evaluation of one-time extensions of containment ILRT intervals. The assessment also followed the guidance and approach outlined in the Indian Point Unit Three Nuclear Power Plant (IP3) ILRT extension submittal and the results and findings from the Pilgrim Probabilistic Safety Assessment (PSA) update are used for this risk assessment.

The Pilgrim Station PSA were used to evaluate the change in population dose rate (person-rem/ry), change in Large Early Release Frequency (LERF), and the change in conditional containment failure probability.

The risk assessment evaluation examined Pilgrim PSA plant specific accident sequences in which the containment integrity remains intact or the containment is impaired. Specifically, the following were considered:

- Core damage sequences in which the containment remains intact initially and in the long term (EPRI Class 1 sequences).
- Core damage sequences in which containment integrity is impaired due to a pre-existing isolation failure of plant components associated with Type A integrated leak rate testing. For example, containment liner breach. (EPRI Class 3 sequences).
- Core damage sequences in which containment integrity is impaired due to pre-existing 'failure-to-seal' failure of plant components associated with either a Type B or Type C local leak rate testing (EPRI Classes 4 and 5 sequences).
- Core damage sequences involving containment isolation failures due to failures-to-close of large containment isolation valves initiated by support system failures, or random or common cause valve failures (EPRI Class 2 sequences) and containment isolation failures of pathways left 'opened' following a plant post-maintenance test, or valve failing to close following a valve stroke test (EPRI Class 6 sequences).
- Core damage sequences involving containment failure induced by severe accident phenomena (EPRI Class 7 sequences) or containment bypassed (EPRI Class 8 sequences).

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

1) Quantify the baseline risk in terms of frequency per reactor year for each of the eight containment release scenario types identified in the EPRI report.

2) Determine the containment leakage rates for applicable cases, 3a and 3b.

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

4) Determine the population dose rate; also know as population dose risk (person-rem/Ry) by multiplying the dose calculated in step (3) by the associated frequency calculated in step (1).

5) Determine the change in probability of leakage detectable only by ILRT, and associated frequency for the new surveillance intervals of interest (Classes 3a and 3b).

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

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

8) Evaluate the risk impact in terms of LERF.

9) Evaluate the change in conditional containment failure probability.

The risk assessment evaluation of the one time ILRT extension is characterized by the following risk metrics: (as used in previously approved ILRT test interval extensions:

- The potential change in population dose rate (person-rem/ry)
- The change in Large Early Release Frequency (LERF)
- The change in conditional containment failure probability (CCFP).

The impact of these risk metrics associated with extending the Type A ILRT interval, are presented in Table ES-1.

The conclusions of the plant internal events risk associated with extending the Type A ILRT interval from ten to fifteen years are as follows.

1) The increase in risk on the total integrated plant risk as measured by person-rem/ry increases for those accident sequences influenced by Type A testing, given the change from a 1-in-10 years test interval to a 1-in-15 years test interval, is found to be 0.009% (0.002 person-rem/ry). This value can be considered to be a negligible increase in risk.

2) Regulatory Guide 1.174 provides guidance for determining the risk impact of plant-specific changes to the licensing basis. Regulatory Guide 1.174 defines very small changes in risk as resulting in increases of core damage frequency (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 1-in-10 years to 1-in-15 years is 1.97×10^{-9}/ry. Since Regulatory Guide 1.174 defines very small changes in LERF as below 10^{-7}/yr, increasing the ILRT interval at Pilgrim from the currently allowed one-in-ten years to one-in-fifteen years is non-risk significant from a risk perspective.

3) The change in conditional containment failure probability (CCFP) is calculated to demonstrate the impact on 'defense-in-depth'. For the current ten-year ILRT interval, sequences involving no containment failure or small releases contribute 1.67% to the overall plant risk. Alternatively stated, the contribution of sequences involving containment failure for the ten-year interval is 98.33%. These numbers are consisted with those documented in the Pilgrim PSA. For the proposed fifteen-year interval, the contribution of sequences involving containment failure increased to 98.36%. Therefore, $)CCFP_{10\text{-}15}$ is found to be 0.03%. This signifies a very small increase and represents a negligible change in the Pilgrim containment defense-in-depth.

In addition to the internal events risk assessment evaluation, the impact associated with extending the Type A test frequency interval is further examined by considering external event hazard or potential containment liner corrosion. The purpose for these additional evaluations is to assess whether there are any unique insights or important quantitative information associated with the explicit consideration of external event hazard or containment liner corrosion in the risk assessment results.

The external event hazards or potential containment liner corrosion evaluation was found not to impact any of the above conclusions. The results from these cases are presented in Tables ES-2 and ES-3 respectively and summarized below.

Considerations of the combined internal events and external event hazards assessment during an extension of the ILRT Interval yielded the following conclusions:

1) Based on conservative methodologies in estimating the combined core damage frequency for internal events, seismic events, and fires events, the increase in LERF from extending the Pilgrim Station ILRT frequency from 1-in-10 years to 1-in-15 years is 1.10×10^{-7}/ry. This value is slightly above the 10^{-7}/yr criterion of Region III, Very Small Change in Risk (Figure 2-1), of the acceptance guidelines in NRC Regulatory Guide 1.174. Consequently, consistent with Regulatory Guide 1.174, the total Pilgrim Station LERF from internal and external events was calculated at 7.30×10^{-6}/ry to demonstrate that LERF is acceptable. This is less than the Regulatory Guide 1.174 acceptance guideline of 10^{-5}/yr (refer to Appendix A). Therefore, increasing the ILRT interval at Pilgrim from the currently allowed 1-in-10 years to 1-in-15 years is non-risk significant from a risk perspective.

2) The combined internal and external events increase in risk on the total integrated plant risk as measured by person-rem/ry increases for those accident sequences influenced by Type A testing, given the change from a 1-in-10 years test interval to a 1-in-15 years test interval, is found to be 0.052% (0.145 person-rem/ry). This value can be considered to be a negligible increase in risk.

3) The change in the combined internal and external events conditional containment failure probability from 1-in-10 years to 1-in-15 years is 0.13%. A change in)CCFP of less than 1% is insignificant from a risk perspective.

4) Other salient results are summarized in Table ES-2. The key results to this risk assessment are those for the 10-year interval (current Pilgrim Station ILRT interval) and the 15-year interval (proposed change).

Recently, the NRC issued a series of Requests for Additional Information (RAIs) in response to the one-time relief requests for the ILRT surveillance interval submitted by various licensees. The RAIs requested a risk analysis on the potential increase in risk due to drywell/torus liner leakage, caused by age-related degradation mechanisms.

The risk analysis utilizes the referenced Calvert Cliffs Nuclear Power Plant assessment to estimate the risk impact from containment liner corrosion during an extension of the ILRT interval. Consistent with the Calvert Cliffs analysis, the following issues were addressed:

- Differences between the containment basemat and the drywell and torus liner
- The historical drywell/torus steel shell 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

Considerations of risk impact of containment liner corrosion during an extension of the ILRT Interval yielded the following conclusions:

1) The impact of including age-adjusted corrosion effects in the ILRT assessment has minimal impact on plant risk and is therefore acceptable.

2) The change in LERF, taking into consideration the likelihood of a containment liner flaw due to age-adjusted corrosion is non-risk significant from a risk perspective. Specifically, extending the interval to 15 years from the current 10 years requirement is estimated to be about 2.47×10^{-9}/ry. This is below the Regulatory Guide 1.174 acceptance criteria threshold of 10^{-7}/yr.

3) The age-adjusted corrosion impact in dose increase is estimated to be 2.70×10^{-3} person-rem/ry or 0.012% from the baseline ILRT 10 year's interval.

4) The age-adjusted corrosion impact on the conditional containment failure probability increase is estimated to be 0.3%.

5) A series of parametric sensitivity studies regarding potential age related corrosion effects on the containment steel liner also demonstrated minimal impact on plant risk.

6) Other salient results are summarized in Table ES-3.

Table ES-1

Internal Events Quantitative Results as a Function of ILRT Interval

EPRI Class	Category Description	Dose (Person-Rem Within 50 miles)[1]	Quantitative Results as a Function of ILRT Interval			
			Current (1-per-10 year ILRT)		Proposed (1-per-15 year ILRT)	
			Accident Frequency (per ry)	Population Dose Rate (Person-Rem / Ry Within 50 miles)	Accident Frequency (per ry)	Population Dose Rate (Person-Rem / Ry Within 50 miles)
1	No Containment Failure [1]	1.06×10^{4}	6.78×10^{-8}	7.20×10^{-4}	4.61×10^{-8}	4.89×10^{-4}
2	Containment Isolation System Failure	4.53×10^{6}	4.42×10^{-11}	2.00×10^{-4}	4.42×10^{-11}	2.00×10^{-4}
3a	Small Pre-Existing Failures [1], [2]	1.06×10^{5}	3.93×10^{-8}	4.17×10^{-3}	5.90×10^{-8}	6.25×10^{-3}
3b	Large Pre-Existing Failures [1], [2]	3.71×10^{5}	3.93×10^{-9}	1.46×10^{-3}	5.90×10^{-9}	2.19×10^{-3}
4	Type B Failures (LLRT)	N/A	0.00	0.00	0.00	0.00
5	Type C Failures (LLRT)	N/A	0.00	0.00	0.00	0.00
6	Other Containment Isolation System Failure	N/A	0.00	0.00	0.00	0.00
7a	Containment Failure Due to Severe Accident (a)[3]	4.53×10^{6}	1.59×10^{-7}	7.20×10^{-1}	1.59×10^{-7}	7.20×10^{-1}
7b	Containment Failure Due to Severe Accident (b) [3]	1.82×10^{6}	2.19×10^{-8}	3.99×10^{-2}	2.19×10^{-8}	3.99×10^{-2}
7c	Containment Failure Due to Severe Accident (c) [3]	4.55×10^{6}	4.38×10^{-6}	1.99×10^{1}	4.38×10^{-6}	1.99×10^{1}
7d	Containment Failure Due to Severe Accident (d) [3]	7.35×10^{5}	1.70×10^{-6}	1.25×10^{0}	1.70×10^{-6}	1.25×10^{0}
8	Containment Bypass Accidents	5.66×10^{6}	3.79×10^{-8}	2.15×10^{-1}	3.79×10^{-8}	2.15×10^{-1}
TOTALS:			6.41×10^{-6}	22.132	6.41×10^{-6}	22.134
Increase in Dose Rate						0.009%
Increase in LERF					1.97×10^{-9}	
Increase in CCFP (%)					0.03%	

Table ES-2

Internal and External Events Quantitative Results as a Function of ILRT Interval

EPRI Class	Category Description	Dose (Person-Rem Within 50 miles)[1]	Quantitative Results as a Function of ILRT Interval			
			Current (1-per-10 year ILRT)		Proposed (1-per-15 year ILRT)	
			Accident Frequency (per ry)	Population Dose Rate (Person-Rem / Ry Within 50 miles)	Accident Frequency (per ry)	Population Dose Rate (Person-Rem / Ry Within 50 miles)
1	No Containment Failure [1]	1.06×10^{4}	7.53×10^{-6}	7.98×10^{-2}	6.32×10^{-6}	6.70×10^{-2}
2	Containment Isolation System Failure	4.53×10^{6}	1.63×10^{-7}	7.38×10^{-1}	1.63×10^{-7}	7.38×10^{-1}
3a	Small Pre-Existing Failures [1], [2]	1.06×10^{5}	2.20×10^{-6}	2.33×10^{-1}	3.30×10^{-6}	3.50×10^{-1}
3b	Large Pre-Existing Failures [1], [2]	3.71×10^{5}	2.20×10^{-7}	8.17×10^{-2}	3.30×10^{-7}	1.22×10^{-1}
4	Type B Failures (LLRT)	N/A	0.00	0.00	0.00	0.00
5	Type C Failures (LLRT)	N/A	0.00	0.00	0.00	0.00
6	Other Containment Isolation System Failure	N/A	0.00	0.00	0.00	0.00
7a	Containment Failure Due to Severe Accident (a)[3]	4.53×10^{6}	6.82×10^{-6}	3.09×10^{1}	6.82×10^{-6}	3.09×10^{1}
7b	Containment Failure Due to Severe Accident (b) [3]	1.82×10^{6}	7.47×10^{-8}	1.36×10^{-1}	7.47×10^{-8}	1.36×10^{-1}
7c	Containment Failure Due to Severe Accident (c) [3]	4.55×10^{6}	4.74×10^{-5}	2.16×10^{2}	4.74×10^{-5}	2.16×10^{2}
7d	Containment Failure Due to Severe Accident (d) [3]	7.35×10^{5}	9.23×10^{-6}	6.79×10^{0}	9.23×10^{-6}	6.79×10^{0}
8	Containment Bypass Accidents	5.66×10^{6}	4.69×10^{-6}	2.66×10^{1}	4.69×10^{-6}	2.66×10^{1}
TOTALS:			7.83×10^{-5}	281.159	7.83×10^{-5}	281.304
Increase in Dose Rate						0.052%
Increase in LERF					1.10×10^{-7}	
Increase in CCFP (%)					0.13%	

Table ES-3

Liner Corrosion Impact Quantitative Results as a Function of ILRT Interval

EPRI Class	Category Description	Dose (Person-Rem Within 50 miles)[1]	Quantitative Results as a Function of ILRT Interval			
			Current (1-per-10 year ILRT)		Proposed (1-per-15 year ILRT)	
			Accident Frequency (per ry)	Population Dose Rate (Person-Rem / Ry Within 50 miles)	Accident Frequency (per ry)	Population Dose Rate (Person-Rem / Ry Within 50 miles)
1	No Containment Failure [1]	1.06×10^{4}	6.76×10^{-8}	7.16×10^{-4}	4.55×10^{-8}	4.83×10^{-4}
2	Containment Isolation System Failure	4.53×10^{6}	4.42×10^{-11}	2.00×10^{-4}	4.42×10^{-11}	2.00×10^{-4}
3a	Small Pre-Existing Failures [1], [2]	1.06×10^{5}	3.91×10^{-8}	4.15×10^{-3}	5.87×10^{-8}	6.22×10^{-3}
3b	Large Pre-Existing Failures [1], [2]	3.71×10^{5}	4.30×10^{-9}	1.59×10^{-3}	6.77×10^{-9}	2.51×10^{-3}
4	Type B Failures (LLRT)	N/A	0.0	0.0	0.0	0.0
5	Type C Failures (LLRT)	N/A	0.0	0.0	0.0	0.0
6	Other Containment Isolation System Failure	N/A	0.0	0.0	0.0	0.0
7a	Containment Failure Due to Severe Accident (a)[3]	4.53×10^{6}	1.59×10^{-7}	7.19×10^{-1}	1.59×10^{-7}	7.19×10^{-1}
7b	Containment Failure Due to Severe Accident (b) [3]	1.82×10^{6}	2.19×10^{-8}	3.99×10^{-2}	2.19×10^{-8}	3.99×10^{-2}
7c	Containment Failure Due to Severe Accident (c) [3]	4.55×10^{6}	4.38×10^{-6}	1.99×10^{1}	4.38×10^{-6}	1.99×10^{1}
7d	Containment Failure Due to Severe Accident (d) [3]	7.35×10^{5}	1.70×10^{-6}	1.25×10^{0}	1.70×10^{-6}	1.25×10^{0}
8	Containment Bypass Accidents	5.66×10^{6}	3.79×10^{-8}	2.15×10^{-1}	3.79×10^{-8}	2.15×10^{-1}
TOTALS:			6.41×10^{-6}	22.1606	6.41×10^{-6}	22.1633
Increase in Dose Rate						0.012%
Increase in LERF					2.47×10^{-9}	
Increase in CCFP (%)					0.30%	

Notes to Tables ES-1, ES-2, and ES-3:

1) Only EPRI classes 1, 3a, and 3b are affected by ILRT (Type A) interval changes.

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

3) EPRI Class 7, containment failure due to severe accident, was subdivided into four subgroups based on Pilgrim Level 2 containment failure modes for dose allocation purposes. Note that this EPRI class is not affected by ILRT interval changes.

Nomenclature

APB	Accident Progression Bin
ATWS	Anticipated Transient Without Scram
CAPB	Collapsed Accident Progression Bin
CCIs	Core-Concrete Interactions
CCFP	Conditional Containment Failure Probability
CD	Core Damage
CDF	Core Damage Frequency
CET	Containment Event Tree
CF	Containment Failure
DCH	Direct Containment Heating
DW	Drywell
EPRI	Electrical Power Research Institute
ILRT	Integrated Leak Rate Testing
IPE	Individual Plant Examination
PEEE	Individual Plant Examination for External Events
ISLOCA	Interface System Loss of Coolant Accident
IP3	Indian Point Unit Three Nuclear Power Plant
LERF	Large Early Release Frequency
LLRT	Local Leak Rate Testing
LOCA	Loss of Coolant Accident
NEI	Nuclear Energy Institute
NRC	United States Nuclear Regulatory Commission
PNPS	Pilgrim Nuclear Power Station
PDS	Plant Damage State

Nomenclature (continued)

PRA	Probabilistic Risk Analysis
PSA	Probabilistic Safety Assessment
RAI	Request for Additional Information
RCS	Reactor Coolant System
RPV	Reactor Pressure Vessel
RF	Refueling Outage
TS	Technical Specifications
WANO	World Association of Nuclear Operations
WW	Wetwell

Definitions

Accident sequence - a representation in terms of an initiating event followed by a combination of system, function and operator failures or successes, of an accident that can lead to undesired consequences, with a specified end state (e.g., core damage or large early release). An accident sequence may contain many unique variations of events (minimal cut sets) that are similar.

Containment event tree – a quantifiable, logical network that begin with a core damage endstate and progresses to possible containment conditions affecting the radionuclide release magnitude and timing.

Core damage - uncovery and heatup of the reactor core to the point at which prolonged oxidation and severe fuel damage is anticipated and involving enough of the core to cause a significant release.

Core damage frequency - expected number of core damage events per unit of time.

Cutsets - Accident sequence failure combinations.

End State - is the set of conditions at the end of an event sequence that characterizes the impact of the sequence on the plant or the environment. End states typically include: success states, core damage sequences, plant damage states for Level 1 sequences, and release categories for Level 2 sequences.

Event tree - a quantifiable, logical network that begins with an initiating event or condition and progresses through a series of branches that represent expected system or operator performance that either succeeds or fails and arrives at either a successful or failed end state.

Initiating Event - An initiating event is any event that perturbs the steady state operation of the plant, if operating, or the steady state operation of the decay heat removal systems during shutdown operations such that a transient is initiated in the plant. Initiating events trigger sequences of events that challenge the plant control and safety systems.

ISLOCA - a LOCA when a breach occurs in a system that interfaces with the RCS, where isolation between the breached system and the RCS fails. An ISLOCA is usually characterized by the over-pressurization of a low-pressure system when subjected to RCS pressure and can result in containment bypass

Large early release - the rapid, unmitigated release of airborne fission products from the containment to the environment occurring before the effective implementation of off-site emergency response and protective actions.

Large early release frequency - expected number of large early releases per unit of time.

Level 1 - identification and quantification of the sequences of events leading to the onset of core damage.

Level 2 - evaluation of containment response to severe accident challenges and quantification of the mechanisms, amounts, and probabilities of subsequent radioactive material releases from the containment.

Plant damage state - Plant damage states are collections of accident sequence end states according to plant conditions at the onset of severe core damage. The plant conditions considered are those that determine the capability of the containment to cope with a severe core damage accident. The plant damage states represent the interface between the Level 1 and Level 2 analyses.

Definitions (continued)

Probability - is a numerical measure of a state of knowledge, a degree of belief, or a state of confidence about the outcome of an event.

Probabilistic risk assessment - a qualitative and quantitative assessment of the risk associated with plant operation and maintenance that is measured in terms of frequency of occurrence of risk metrics, such as core damage or a radioactive material release and its effects on the health of the public (also referred to as a probabilistic safety assessment, PSA).

Release category - radiological source term for a given accident sequence that consists of the release fractions for various radionuclide groups (presented as fractions of initial core inventory), and the timing, elevation, and energy of release. The factors addressed in the definition of the release categories include the response of the containment structure, timing, and mode of containment failure; timing, magnitude, and mix of any releases of radioactive material; thermal energy of release; and key factors affecting deposition and filtration of radionuclides. Release categories can be considered the end states of the Level 2 portion of a PSA.

Risk - encompasses what can happen (scenario), its likelihood (probability), and its level of damage (consequences).

Risk metrics - the quantitative value, obtained from a PRA analysis, used to evaluate the results of an application (e.g., CDF or LERF).

Severe accident - an accident that involves extensive core damage and fission product release into the reactor vessel and containment, with potential release to the environment.

Split Fraction - a unitless parameter (i.e., probability) used in quantifying an event tree. It represents the fraction of the time that each possible outcome, or branch, of a particular top event may be expected to occur. Split fractions are, in general, conditional on precursor events. At any branch point, the sum of all the split fractions representing possible outcomes should be unity. (Popular usage equates "split fraction" with the failure probability at any branch [a node] in the event tree.)

Vessel Breach - a failure of the reactor vessel occurring during core melt (e.g., at a penetration or due to thermal attack of the vessel bottom head or wall by molten core debris).

SECTION 1

INTRODUCTION

1.1 Purpose

The purpose of this report is to provide supplemental information to support the proposed Pilgrim Nulcear Power Station (Pilgrim Station) Technical Specifications (TS) change of implementing a one-time extension of the containment Type A Integrated Leak Rate Test (ILRT) interval from ten years to fifteen years.

The risk assessment follows the guidelines from NEI 94-01 "Industry Guideline for Implementing Performance-Based Option of 10 CFR Part 50, Appendix J" [1], the methodology used in EPRI TR-104285 "Risk Assessment of Revised Containment Leak Rate Testing Intervals" [3] and the guidance provided in NRC Regulatory Guide 1.174, "An Approach for Using Probabilistic Risk Assessment In Risk-Informed Decisions On Plant-Specific Changes to the Licensing Basis" [6]. The assessment also followed the guidance and additional information distributed by NEI in November 2001 to their Administrative Points of Contact regarding risk assessment evaluation of one-time extensions of containment ILRT intervals [4 & 5]. The assessment also followed the guidance and approach outlined in the Indian Point Unit Three Nuclear Power Plant (IP3) ILRT extension submittal [8] and the results and findings from the Pilgrim Probabilistic Safety Assessment (PSA) update [7] are used for this risk assessment.

1.2 Background

In October 26, 1995, the Nuclear Regulatory Commission (NRC) revised 10 CFR 50, Appendix J. The revisions to Appendix J allow licensees to choose containment leakage testing under Option A "Prescriptive Requirements" or Option B "Performance-Based Requirements," for leakage-rate testing of light-water-cooled containments.

The adoption of the Option B performance-based containment leakage rate-testing program did not alter the basic method by which Appendix J leakage rate testing is performed, but did alter the frequency of measuring primary containment leakage in Type A, B and C tests. Frequency is based upon an evaluation which looks at the "as found" leakage history to determine a frequency for leakage testing which provides assurance that leakage limits will be maintained. The changes to Type A test frequency allowed by Option B do not directly result in an increase in containment leakage, only the interval at which such leakage is measured on an integrated basis.

Under Option B, the ILRT Type A surveillance testing requirements was extended from three-in-ten years to at least once per ten years. The revised Type A test 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 is less than the maximum allowable containment leakage limit of $1.0L_a$.

In accordance with the revised containment leakage-rate testing for Appendix J, the Pilgrim Station selected the requirements under Option B as its testing program. Pilgrim Station current ten-year Type A test is due to be performed during refueling outage fifteen (RFO 15), scheduled for April/May 2005. However, Pilgrim Station seeks a one-time exemption based on the substantial cost savings of removing 2 days of critical path time from RFO 15 and therefore allows deferral of the associated costs out to RFO 17 in 2009. In addition, this initiative directly supports site goals related to capacity factor and World

Association of Nuclear Operators (WANO) performance by shortening planned outage duration for RFO 15.

The basis for the current 10-year test interval is provided in NEI 94-01, Revision 0, (Section 11.0) which was issued in 1995 during development of the performance-based Option B to Appendix J [1]. This document is based upon a generic evaluation documented in NUREG-1493, "Performance-Based Containment Leak-Test Program", [2] as the technical basis to support regulatory rulemaking in revising the testing requirements to Appendix J, Option B.

The NUREG-1493 [2] report examined the impact of containment leakage on public health and safety associated with a range of extended leakage rate test intervals.

NUREG-1493 made the following observations with regard to extending the test frequency:

- *"Reducing the Type A (ILRT) testing frequency to one per twenty years was found to lead to an imperceptible increase in risk. The estimated increase in risk is small because ILRTs identify only a few potential leakage paths that cannot be identified by Type B and C testing, and the leaks that have been found by Type A tests have been only marginally above the existing requirements. Given the insensitivity of risk to containment leakage rate, and the same fraction of leakage detected solely by Type A testing, increasing the interval between ILRT testing had minimal impact on public risk."*
- While Type B and C tests identify the vast majority (greater than 95%) of all potential leakage paths, performance-based alternatives are feasible without significant risk impacts. Since leakage contributes less than 0.1 percent of overall risk under existing requirements, the overall effect is very small.

NUREG-1493 analyzed both Boiling Water Reactors (Peach Bottom and Grand Gulf) and Pressurized Water Reactors (Surry, Sequoyah, and Zion). For Peach Bottom, (a comparable Boiling Water Reactor plant to Pilgrim), it was found that increasing the containment leak rates several orders of magnitude over the design basis (0.5 percent per day to 50 percent per day), results in a negligible increase in total population exposure. Therefore, extending the ILRT interval does not result in any significant increase in risk.

To supplement the NRC's rulemaking basis, NEI undertook another similar study. The results of that study are documented in EPRI research project report TR-104285 [3]. The EPRI TR-104285 study combined PSA Level 2[1] models with NUREG-1150 "Severe Accident Risks: An Assessment for Five U.S. Nuclear Power Plants" [9] Level 3[2] population dose models to perform the analysis. This study also used the approach of NUREG-1493 [2] in calculating the increase in pre-existing leakage probability due to extending the ILRT and LLRT test intervals. The EPRI Methodology [3] used a simplified risk model--PRA containment event trees (CETs). These CETs provide a risk framework for evaluating the effect of containment isolation 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

[1] Level 2 - the evaluation of containment response to severe accident challenges and quantification of the mechanisms, amounts, and probabilities of subsequent radioactive material releases from the containment.

[2] Level 3 - A measure of containment failure sequences leading to public health effects and their frequencies.

reviewed in the study into eight classifications to permit the appropriate delineation among containment isolation failure and containment failure due severe accident phenomena. The eight EPRI accident classes in the simplified model are:

1) Containment remains intact initially and in the long term. The release of fission products (and accident consequences) is determined by the maximum allowable containment leakage.

2) Core damage accident sequences in which containment integrity is impaired due independent (or random) containment isolation failures that include those accident s sequences in which the containment isolation system function fails during the accident progression (i.e., failures-to-close of large containment isolation valves initiated by support system failures, or random or common cause valve failures).

3) Core damage sequences in which containment integrity is impaired due to a pre-existing isolation failure of plant components associated with Type A integrated leak rate testing. For example, containment liner breach.

4) Core damage sequences in which containment integrity is impaired due to an independent (or random) pre-existing isolation failure-to-seal of plant components associated with Type B integrated leak rate testing. These are the Type B-tested components that have isolated but exhibit excessive leakage.

5) Core damage sequences in which containment integrity is impaired due to an independent (or random) pre-existing isolation failure-to-seal of plant components associated with Type C integrated leak rate testing.

6) Core damage sequences in which containment integrity is impaired due to containment isolation failures that include those leak paths not identified by containment leak rate tests. The type of failures considered under this Class includes those valves left open or valves that did not properly seal following test or maintenance activities.

7) Core damage sequences involving containment failure induced by severe accident phenomena. Changes in ILRTs or LLRTs requirements do not impact these accidents.

8) Core damage sequences in which the containment is bypassed (either as an initial condition or induced by accident phenomena). Changes in ILRTs or LLRTs requirements do not impact these accidents.

These eight accident classes allow the isolation failures modes and type of penetration analyzed to be correlated directly with Types A, B, and C test relaxation benefits. Each of the eight classes was categorized according to certain release characterization to determine the baseline incremental risk.

Building upon the methodology of the EPRI TR-104285 [3] study, the Indian Point Unit Three (IP3) Methodology [8], quantified leakage from accident sequences in endstate 3 (reclassified as 3a and 3b). Accident sequence endstates 3a and 3b have the potential to result in a change in risk associated with changes in ILRT intervals since a pre-existing leak is assumed to be present for these endstates. By manipulating the probability of a pre-existing leak of sufficient leak size, an evaluation of the change in large early release frequency (LERF) can be performed. The NRC [10] considered this an improvement on the EPRI study [3]. Similar information is contained in the Crystal River Nuclear Power Plant submittal [11].

Based on the improved methodology, NEI issued in November 2001enhanced guidance "Interim Guidance for Performing Risk Impact Assessments In Support of One-Time Extensions for Containment Integrated Leakage Rate Test Surveillance Intervals" [4], and "Additional Information for ILRT Extensions," [5] that builds on the EPRI TR-104285 [3], IP3 [8] and Crystal River submittal [11] methodology and is intended to provide for more consistent submittals to the NRC.

The Pilgrim Station evaluation assesses the change in the predicted population dose rate associated with the interval extension. The assessment also evaluated the risk increase resulting from extending the ILRT interval in terms of Large Early Release Frequency (LERF), and the impact on Conditional Containment Failure Probability (CCFP). Regulatory Guide 1.174 [6] provides guidance for using PRA in risk-informed decisions for determining the risk impact of plant-specific changes to the licensing basis. Regulatory Guide 1.174 [6] defines very small changes in the risk acceptance guidelines as increases in Core Damage Frequency (CDF) of less than 10^{-6} per reactor year and increases in LERF of less than 10^{-7} per reactor year. Since the Type A test does not impact CDF, the only relevant criterion is the change in LERF. Regulatory Guide 1.174 [6] also encourages the use of risk analysis techniques to help ensure and demonstrate that key risk metrics such as defense-in-depth philosophy, are satisfied. Based on that, the increase in the CCFP, which helps to ensure that the defense-in-depth philosophy is maintained, was evaluated.

SECTION 2

EVALUATION

2.1 Method of Analysis

The Pilgrim Station risk assessment was performed following the guidelines of NEI 94-01, "Industry Guideline for Implementing Performance-Based Option of 10 CFR Part 50, Appendix J" [1], the methodology used in EPRI TR-104285, "Risk Assessment of Revised Containment Leak Rate Testing Intervals," [3] and the guidance provided in NRC Regulatory Guide 1.174, "An Approach for Using Probabilistic Risk Assessment In Risk-Informed Decisions On Plant-Specific Changes to the Licensing Basis" [6]. The assessment also followed the guidance and additional information distributed by NEI in November 2001 to their Administrative Points of Contact regarding risk assessment evaluation of one-time extensions of containment ILRT intervals [4 & 5]. The Pilgrim Station risk assessment also followed the guidance and approach outlined in the Indian Point Unit Three Nuclear Power Plant (IP3) ILRT extension submittal [8] and the results and findings from the Pilgrim Probabilistic Safety Assessment (PSA) update [7] are used for this risk assessment.

Consistent with the NEI interim guidance [4, 5], the Pilgrim Station risk impact assessment of extending containment Type A test interval involves a nine-step process as follows:

1) Quantify the baseline risk in terms of frequency per reactor year for each of the eight containment release scenario types identified in the EPRI report.

2) Determine the containment leakage rates for applicable cases, 3a and 3b.

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

4) Determine the population dose rate; also know as population dose risk (person-rem/ry) by multiplying the dose calculated in step (3) by the associated frequency calculated in step (1).

5) Determine the change in probability of leakage detectable only by ILRT, and associated frequency for the new surveillance intervals of interest (Classes 3a and 3b). Note that with increases in the ILRT surveillance interval, the size of the postulated leak path and the associated leakage rate are assumed not to change, however the probability of leakage detectable only by ILRT does increase.

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

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

8) Evaluate the risk impact in terms of LERF.

9) Evaluate the change in conditional containment failure probability.

The first seven steps of the methodology calculate the change in dose. The change in dose is the primary basis upon which the Type A ILRT interval extension was previously granted for IP3 [8, 10] and other subsequent extensions [11].

The eighth step in the interim methodology calculates the change in LERF and compares it to the guidelines in Regulatory Guide 1.174 [6]. Because the change in ILRT test interval does not impact the

CDF, the relevant criterion is LERF. The final step of NEI's interim methodology calculates the change in containment failure probability given the change of ILRT test interval from once-per-10 years to once-per-15 years.

2.2 Assumptions

1) The surveillance frequency for Type A testing in NEI 94-01 [1] is at least once per ten years based on an acceptable performance history. Based on the consecutive successful ILRTs performed in the early 1990's, the current ILRT interval for Pilgrim Station is once per ten years [13].

2) The Pilgrim Station (Revision 1) Level 1[3] and Level 2 internal events IPE models provide representative results for the analysis [7].

3) Radionuclide release categories defined in this report are consistent with the EPRI TR-104285 methodology. [3]

4) The EPRI methodology concluded that Severe Accident Phenomena and Bypass Classes accident sequences (e.g., drywell liner melt-through, ATWS or Interface system LOCA, ISLOCA) contribution to population dose is unchanged by the proposed ILRT extension. These Classes are included for comparison purposes. As such, no changes in this analysis will alter this conclusion.

5) The reliability of containment isolation valves to close in response to a containment isolation signal is not impacted by the change in ILRT frequency.

6) The maximum containment leakage for Class 1 sequences is 1La [3]. (La is the Technical Specification maximum allowable containment leakage rate).

7) The maximum containment leakage for Class 3a sequences per the NEI Interim Guidance [4] and previously approved methodology [8, 10] is 10La.

8) The maximum containment leakage for Class 3b sequences per the NEI Interim Guidance [3] and previously approved methodology [8, 10] is 35La.

9) Class 3b release is categorized as LERF, based on the previously approved IP3 ILRT extension [8, 10] and NEI's interim methodology [4].

10) Containment leak rates greater than 2La but less than 35La indicate an impaired containment. The leak rate is considered 'small' per the NEI Interim Guidance [4] and previously approved methodology [3, 8, and 10]. Furthermore, these releases have a break opening of greater than 0.5-inch but less than 2-inch diameter [8, 10].

11) Containment leak rates greater than 35La indicates a containment breach. This leak rate is considered 'large' per the NEI Interim Guidance [4] and previously approved methodology [8, 10].

[3] Level 1 - identification and quantification of the sequences of events leading to the onset of core damage.

12) Containment leak rates less than 2La indicates an intact containment. This leak rate is considered as 'negligible' per the NEI Interim Guidance [4] and previously approved methodology [8, 10].

13) EPRI accident Class 2 (Large Containment Isolation Failures) potential releases can be consider similar to a release associated with early drywell failure at high reactor pressure vessel (RPV) pressure.

14) Because EPRI Class 8 sequences are containment bypass sequences, potential releases are directly to the environment. Therefore, the containment structure will not impact the release magnitude.

15) An evaluation of the risk impact of the ILRT on shutdown risk is addressed using the generic results from EPRI TR-104285 [3] as augmented by NEI Interim Guidance [4, 5].

2.3 Data and Design Criteria

1) The Pilgrim Station Level 1 and 2 PSA update is used as input to this analysis reflects the as built, as-operated plant. [7]

2) The point estimate CDF value, as reported in the Pilgrim Station PSA, Revision 1 is 6.41×10^{-6}/ry[4]. [7]

3) The Pilgrim Station Level 2 PSA [7] is used to calculate the release frequencies for the accidents evaluated in this assessment. Table 2-1 summarizes the Pilgrim Station Level 1 PSA internal events point estimate frequency results by core damage accident class.

4) The pertinent Pilgrim Station Level 2 PSA results for containment failure is summarized in Table 2-2.

5) 4 The total LERF for Pilgrim is 1.13×10^{-7}/ry [7]. This frequency is the frequency that results from internal causes and applies to the plant as it is currently configured and operated. Six types of accidents dominate the internal large early release: accidents initiated by station blackout, anticipated transient without scram, transients, interfacing system loss of coolant accidents, loss-of-coolant accidents and vessel rupture events. Their point estimate contributions to the total internal large early release frequency are listed in Table 2-3.

6) The pertinent Pilgrim Station Level 2 PSA results in terms of containment release modes are summarized in Table 2-4. The total release frequency is 6.30×10^{-6}/ry; with a total CDF of 6.41×10^{-6}/ry. The containment release modes are listed in the following form: no containment failure (CAPB-1 to CAPB-3), early torus failure (CAPB-4 to CAPB-7), early drywell failure, (CAPB-8 to CAPB-11) late torus failure (CAPB-12 and CAPB-13), late drywell failure (CAPB-14 and CAPB-15) and containment bypass (CAPB-16 to CAPB-19).

7) The random large containment isolation failure probability, from the Pilgrim Station PSA, Revision 1, Section 4.11 [7] is = 6.9×10^{-6} [frequency of containment isolation failure (4.42×10^{-11}) / point estimate CDF (6.41×10^{-6})].

[4] The Level 2 analysis used a point estimate CDF of 6.41 x 10-6/ry. Therefore, this analysis uses the point estimate CDF value in calculating the eight accident classes' frequencies.

8) The conditional failure probability of having a small pre-existing containment leak is 0. 027. This value is based on work performed in the IP3 ILRT submittal [8] and the NEI Interim Guidance [4]. From the IP3 submittal, the probability that a liner leak will be small made use of the data presented in NUREG-1493 [2]. The data reported in NUREG-1493 found that 23 of 144 tests had allowable leak rates in excess of 1.0La. However, of these 23 'failures' only 4 were found by an Type A ILRT, the others were found by Type B and C testing or errors in test alignments. Therefore, the number of failures considered for 'small releases' are 4-of-144. Recent data collected by NEI and documented in the NEI Interim Guidance [4] found that an additional 38 ILRT have been performed since 1/1/95, with only one failure occurring. This indicates a failure probability of 5/182 (0.027) for a type A ILRT.

9) The conditional failure probability of having a large pre-existing containment leak is 0.0027. This value is derived from the NEI Interim Guidance [4]. It's based on the Jeffreys non-informative prior distribution[5] for zero failures. The formula is as follows:

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

The number of large failures is zero, so the probability is 0.5/183=0.0027.

[5] Application of the Jeffreys non-informative prior is one of a number of statistical analysis approaches to estimating probabilities when no failures have been experienced. The approach was used in NUREG-1150 and more recently in NUREG/CR-5750. NUREG/CR-5750 is now the preferred source of initiating event data, which also involves rare event approximations. The selected approach is more conservative than many other statistical approaches.

2.4 Internal Events Impact

This section provides a step-by-step summary of the NEI guidance [4] as applied to the Pilgrim Nuclear Power Station ILRT interval extension risk assessment. Each subsection addresses a step in the NEI guideline [4].

2.4.1 Quantify Baseline Accident Classes Frequencies (Step 1)

This step involves the quantification of the baseline frequencies for each of the EPRI TR-104285 accident classes [3].

Frequency of EPRI Class 1 Sequences. This group consists of all core damage accident progression sequences in which the containment remains isolated and intact (or containment leakage at or below maximum allowable Technical Specification leakage).

Consistent with NEI Interim Guidance [4], the frequency per reactor year for these sequences is calculated by subtracting the frequencies of EPRI Classes 3a and 3b from the sum of all severe accident progression sequence frequencies in which the containment is isolated and intact:

$$\text{CLASS\_1\_FREQUENCY} = \text{NCF} - \text{CLASS\_3a\_FREQUENCY} - \text{CLASS\_3b\_FREQUENCY}$$

Where:

CLASS_1_FREQUENCY = frequency of EPRI Class 1 given a 3-in-10 years ILRT interval

NCF = frequency in which containment leakage is at or below maximum allowable Technical Specification leakage
= 1.11×10^{-7} /ry [Table 2-2]

CLASS_3a_FREQUENCY = frequency of small pre-existing containment liner leakage
= 1.18×10^{-8}/ry [See below write-up]

CLASS_3b_FREQUENCY = frequency of large pre-existing containment liner leakage
= 1.18×10^{-9}/ry [See below write-up]

Therefore:

$$\text{CLASS\_1\_FREQUENCY} = 1.11 \times 10^{-7} - 1.18 \times 10^{-8} - 1.18 \times 10^{-9}$$

$$\text{CLASS\_1\_FREQUENCY} = 9.81 \times 10^{-8}/\text{ry}$$

Frequency of EPRI Class 2 Sequences. This group consists of all core damage accident progression bins in which the containment isolation system function fails during the accident progression. These sequences are dominated by failure-to-close of large (>2-inch diameter) containment isolation valves [6]. The frequency per reactor year for these sequences is determined as follows:

$$\text{CLASS\_2\_FREQUENCY} = \text{PROB}_{\text{large CI}} * \text{CDF}$$

Where:

CLASS_2_FREQUENCY = frequency of EPRI Class 2 given a 3-in-10 years ILRT interval

$PROB_{large\ CI}$ = random large containment isolation failure probability (i.e. large valves)
= 6.9×10^{-6} [Section 2.3, input#7]

CDF = Pilgrim Station PSA core damage frequency = 6.41×10^{-6}/ry` [Section 2.3, input #2]

Therefore:

$$CLASS\_2\_FREQUENCY = 6.9 \times 10^{-6} * 6.41 \times 10^{-6}$$

$$CLASS\_2\_FREQUENCY = 4.42 \times 10^{-11}/ry$$

Frequency of EPRI Class 3a Sequences. This group consists of all core damage accident progression bins for which a small pre-existing leakage in the containment structure (i.e. containment liner) exists. This type of failure is identifiable only from an ILRT and therefore, affected by a change in ILRT testing frequency.

Consistent with NEI Interim Guidance [5], the frequency per reactor year for this category is calculated as the pre-existing leakage probability multiplied by the residual CDF determined as the total CDF minus the CDF for those individual sequences that either may already (independently) cause a LERF or could never cause a LERF:

$$CLASS\_3a\_FREQUENCY = PROB_{class\_3a} * [CDF - (CDF_{LERF} + CDF_{NO\_LERF})]$$

Where:

CLASS_3a_FREQUENCY = frequency of EPRI Class 3a given a 3-in-10 years ILRT interval

$PROB_{class\_3a}$ = probability of small pre-existing containment liner leakage
= 0.027 [Section 2.3, input#8]

CDF = Pilgrim Station PSA core damage frequency = 6.41×10^{-6}/ry [Section 2.3, input#2]

CDF_{LERF} = CDF for those individual sequences that independently cause a LERF. This is denoted from the following accident sequences [Table 2-3]:

- Station Blackout = 6.43×10^{-8}/ry
- Anticipated Transient without Scram = 4.49×10^{-8}/ry
- Transients = 2.26×10^{-9}/ry
- Interfacing System LOCAs = 1.27×10^{-9}/ry
- LOCAs = 1.47×10^{-11}/ry
- Vessel Rupture = 7.91×10^{-12}/ry

$= 6.43 \times 10^{-8}/ry + 4.49 \times 10^{-8}/ry + 2.26 \times 10^{-9}/ry + 1.27 \times 10^{-9}/ry + 1.47 \times 10^{-11}/ry + 7.91 \times 0^{-12}/ry$
$= 1.13 \times 10^{-7}/ry$

CDF_{NO_LERF} = CDF for those individual sequences that never cause a LERF. This is denoted from the loss of containment heat removal accident sequences (Pilgrim Station Class II)
= 5.86×10^{-6}/ry [Table 2-1]

Therefore,

$$CLASS\_3a\_FREQUENCY = 0.027 * [6.41 \times 10^{-6}/ry - (1.13 \times 10^{-7}/ry + 5.86 \times 10^{-6}/ry)]$$

$$CLASS\_3a\_FREQUENCY = 1.18 \times 10^{-8}/ry$$

Frequency of EPRI Class 3b Sequences. This group consists of all core damage accident progression bins for which a large pre-existing leakage in the containment structure (i.e. containment liner) exists. This type of failure is identifiable only from an ILRT and therefore, affected by a change in ILRT testing frequency.

Consistent with NEI Interim Guidance [5], the frequency per reactor year for this category is calculated as:

$$CLASS\_3b\_FREQUENCY = PROB_{class\_3b} \times [CDF - (CDF_{LERF} + CDF_{NO\_LERF})]$$

Where:

CLASS_3b_FREQUENCY = frequency of EPRI Class 3b given a 3-in-10 years ILRT interval

$PROB_{class\_3b}$ = probability of large pre-existing containment liner leakage
= 0.0027 [Section 2.3, input #9]

CDF = Pilgrim Station PSA core damage frequency = 6.41×10^{-6} ry [Section 2.3, input #2]

Therefore,

$$CLASS\_3b\_FREQUENCY = 0.0027 * [6.41 \times 10^{-6}/ry - (1.13 \times 10^{-7}/ry + 5.86 \times 10^{-6}/ry)]$$

$$CLASS\_3b\_FREQUENCY = 1.18 \times 10^{-9}/ry$$

Frequency of EPRI Class 4 Sequences. This group consists of all core damage accident progression sequences in which the containment isolation system function fails due to a pre-existing failure-to-seal of Type B test component(s). Consistent with NEI Interim Guidance [4], because these failures are detected by Type B tests and not by the Type A ILRT, this group is not evaluated further in this analysis.

Frequency of EPRI Class 5 Sequences. This group consists of all core damage accident progression sequences in which the containment isolation system function fails due to a pre-existing failure-to-seal of Type C test component(s). Consistent with NEI Interim Guidance [4], because these failures are detected by Type C tests, this group is not evaluated any further.

Frequency of EPRI Class 6 Sequences. This group consists of all core damage accident sequences in which the containment isolation function is failed due to "other" pre-existing failure modes (e.g., pathways left open or misalignment of containment isolation vales following a test/maintenance evolution). Consistent with NEI Interim Guidance [4], because these failures are detected by Type B or C tests, this group is not evaluated any further.

Frequency of EPRI Class 7 Sequences. This group consists of all core damage accident progression bins in which containment failure induced by severe accident phenomena occurs (i.e. liner melt-through). Consistent with NEI Interim Guidance [4], the frequency per reactor year for this class is based on the plant Level 2 PSA results.

Because the Pilgrim PSA IPE Level 2 containment failure results are summarized into four different release bins (Table 2-2), EPRI Class 7 is sub-divided in this report to reflect this sub-division of the Pilgrim Station Level PSA 2 results. The following sub-classes are defined:

- Class 7a: severe accident induced early drywell failures resulting in early high magnitude releases.

- Class 7b: severe accident induced early torus failures resulting in early medium high or early medium low releases.
- Class 7c: severe accident induced late drywell failures resulting in late high magnitude releases.
- Class 7d: severe accident induced early torus failures resulting in late medium high or late medium low releases.

The frequency of Category 7a is the total frequency of the Pilgrim Station Level 2 PSA early drywell failures release bins (CAPB-8, CAPB-9, CAPB-10 and CAPB-11). Based on the Pilgrim Station Level 2 PSA results summarized in Table 2-4, the frequency of Category 7a is 1.59×10^{-7}/ry.

The frequency of Category 7b is the total frequency of the Pilgrim Station Level 2 PSA early torus failures release bins (CAPB-4, CAPB-5, CAPB-6 and CAPB-7). Based on the Pilgrim Station Level 2 PSA results summarized earlier in Table 2-4, the frequency of Category 7b is 2.19×10^{-8}/ry.

The frequency of Category 7c is the total frequency of the Pilgrim Station Level 2 PSA late drywell failures release bins (CAPB-14 and CAPB-15). Based on the Pilgrim Station Level 2 PSA results summarized earlier in Table 2-4, the frequency of Category 7c is 4.38×10^{-6}/ry.

The frequency of Category 7d is the total frequency of the Pilgrim Station Level 2 PSA late torus failures release bins (CAPB-12 and CAPB-13). Based on the Pilgrim Station Level 2 PSA results summarized earlier in Table 2-4, the frequency of Category 7d is 1.70×10^{-6}/ry.

<u>Frequency of EPRI Class 8 Sequences.</u> This group consists of all core damage accident progression bins in which the accident is initiated by a containment bypass scenario (i.e., ATWS with high power oscillations or Interfacing Systems LOCA). Based on the Pilgrim Station Level 1 PSA results summarized earlier in Table 2-1, the frequency of Classes IV and V is 3.79×10^{-8}/ry.

Note: for EPRI class 8 the maximum release is not based on the maximum allowable containment leakage, because the releases are released directly to the environment. Therefore, the containment structure will not impact the release magnitude.

The EPRI TR-104285 Class frequencies that result in radionuclide releases to the public are derived in accordance with NEI Interim Guidance [4]. The EPRI TR-104285 Class accident sequence frequency results are summarized in Table 2-5.

2.4.2 Containment Leakage Rates (Step 2)

This step defines the containment leakage rates for EPRI accident Classes 3a and 3b. As defined in Step 1, accident Class 3a and 3b are plant accidents with pre-existing containment leakage pathways (designated as "small" and "large") that are identifiable only when performing a Type A ILRT.

The NEI Interim Guidance [4] recommends containment leakage rates of 10La and 35La for accident Classes 3a and 3B, respectively. These values are consistent with previous ILRT frequency extension submittal applications [8]. La is the plant Technical Specification maximum allowable containment leak rate; for Pilgrim La is1.0% of containment air weight per day (per Pilgrim Station Technical Specification).

By definition, and per the NEI Interim Guidance [4] and previously approved methodology [8] the containment leakage rate for Class 1 (i.e., accidents with containment leakage at or below maximum allowable Technical Specification leakage) is 1 La.

2.4.3 Baseline Population Dose Estimate (Step 3)

This step estimates the baseline population dose (person-rem) for each of the EPRI TR-104285 accident classes [3]. The NEI Interim Guidance [4] recommends two options for calculating population dose for the EPRI accident classes:

- Use of NUREG-1150 dose calculations [9]
- Use of plant-specific dose calculations

Because the Pilgrim Station has a Level 3 PSA [7, & 12] and associated plant-specific dose, this risk assessment uses plant specific dose results.

The Pilgrim Station PSA offsite consequences are calculated by the MACCS2 consequence model [12]. The principal phenomena analyzed are atmospheric transport of radionuclides, mitigative actions (i.e., evacuation, condemnation of contaminated crops and milk) based on dose projection, dose accumulation by a number of pathways, including food and water ingestion and economic costs. Input for the Level 3 analysis includes Pilgrim core radionuclide inventory, source terms from the Level 2 (containment performance analysis) model, site metrological data, projected population distribution (within 50-mile radius) for the year 2025, emergency response evacuation modeling and economic data.

The Pilgrim Station consequence analysis looks at the source term for nineteen collapsed accident progression bins (Table 2-4). These bins represent the source term for each of the seventy-seven different containment release modes associated with endstates of the Pilgrim containment event tree (Section 4.7 of Reference 7).

The MACCS2 code was used to estimate the consequences in terms of population dose within 50-miles and offsite economic cost. The Pilgrim Station Level 3 PSA MACCS2 population dose results are presented in Table 2-6. (Use of dose results for the 50-mile radius around the plant, as a figure of merit in this risk evaluation is consistent with NUREG-1150 [9], past ILRT [8 &11] frequency extension submittals, and the NEI Interim Guidance. [4 & 5]).

The Pilgrim Station populations dose information presented in Table 2-6 when combined with the preceding information on the EPRI TR-104285 Class accident sequence frequency results (Table 2-5), provides the basis for the assignment of population dose for each EPRI accident category.

<u>Population Dose for EPRI Class 1</u>. The dose for the "no containment failure" EPRI class 1 sequences is based on collapsed accident progression bin-3 (core damage occurs followed by vessel breach. The containment does not fail structurally and is not vented. However, ex-vessel releases are not recovered in time, and therefore core-concrete interactions occur). Therefore,

$$\text{CLASS\_1\_DOSE} = 1.06 \times 10^2 \text{ person-sv} * \frac{100 \text{ person-rem}}{1 \text{person-sv}} \qquad \text{[Table 2-6]}$$

$$= 1.06 \times 10^4 \text{ person-rem}$$

<u>Population Dose for EPRI Class 2</u>. The 50-miles population dose for the EPRI accident Class 2 (Large Containment Isolation Failures, failure-to-close) is based on the Pilgrim Station collapsed accident progression bin 10 (Table 2-6) as the one closest to the definition of large containment isolation failure.

This selection is based on assuming that the containment isolation failure of EPRI accident Class 2 occurs concurrent with early drywell failure at high RPV pressure. Collapsed accident progression bin 10 results in the highest dose of all of the Pilgrim Station "containment failure" collapsed accident progression bins (which is indicative of an early drywell containment failure with torus pool bypass and extensive core-concrete interactions). Therefore,

$$\text{CLASS\_2\_DOSE} = 4.53 \times 10^4 \text{ person-sv} * \frac{100 \text{ person-rem}}{1 \text{person-sv}} \qquad \text{[Table 2-6]}$$

$$= 4.53 \times 10^6 \text{ person-rem}$$

Population Dose for EPRI Class 3. The 50-miles population dose for the EPRI accident Class 3a (Small Isolation Failures-Liner breach) and accident Class 3b (Large Isolation Failures-Liner breach), per the NEI Interim Guidance [4], are taken as factors of 10La and 35La [4, 8], respectively, times the population dose of EPRI accident Class 1. Therefore,

$$\text{CLASS\_3a\_DOSE} = 10 * \text{CLASS\_1\_DOSE}$$
$$\text{CLASS\_3b\_DOSE} = 35 * \text{CLASS\_1\_DOSE}$$

$$\text{CLASS\_3a\_DOSE} = 10 * 1.06 \times 10^4 \text{ person-rem}$$
$$\text{CLASS\_3b\_DOSE} = 35 * 1.06 \times 10^4 \text{ person-rem}$$

$$\text{CLASS\_3a\_DOSE} = 1.06 \times 10^5 \text{ person-rem}$$
$$\text{CLASS\_3b\_DOSE} = 3.71 \times 10^5 \text{ person-rem}$$

Population Dose for EPRI Classes 3, 4, 5 &6. Per the NEI Interim Guidance [4], EPRI accident Classes 4 (Small Isolation Failure - failure-to-seal, Type B test), 5 (Small Isolation Failure - failure-to-seal, Type C test), and 6 (Containment Isolation Failures, dependent failures, personnel errors) are not affected by ILRT frequency and are not analyzed as part of this risk assessment. Therefore no selections of population does estimates are made for these accident classes.

Population Dose for EPRI Class 7a. The 50-miles population dose for the EPRI accident Class 7a (Severe Accident Phenomena Induced Early Drywell Failures) is based on the Pilgrim Station collapsed accident progression bin 10 (early drywell containment failure with torus pool bypass and extensive core-concrete interactions) as the ones closest to the definition of early drywell failure. Therefore,

$$\text{CLASS\_7a\_DOSE} = 4.53 \times 10^4 \text{ person-sv} * \frac{100 \text{ person-rem}}{1 \text{person-sv}} \qquad \text{[Table 2-6]}$$

$$= 4.53 \times 10^6 \text{ person-rem}$$

Population Dose for EPRI Class 7b. The 50-miles population dose for the EPRI accident Class 7b (Severe Accident Phenomena Induced Early Torus Failures) is based on the Pilgrim Station collapsed accident progression bin 5 (early torus containment failure with drywell floor flooded because of an overlaying pool of water) as the ones closest to the definition of early torus failures. Therefore,

$$\text{CLASS\_7b\_DOSE} = 1.82 \times 10^4 \text{ person-sv} * \frac{100 \text{ person-rem}}{1 \text{person-sv}} \qquad \text{[Table 2-6]}$$

$$= 1.82 \times 10^6 \text{ person-rem}$$

Population Dose for EPRI Class 7c. The 50-miles population dose for the EPRI accident Class 7c (Severe Accident Phenomena Induced Late Drywell Failures) is based on the Pilgrim Station collapsed accident progression bin 15 (Table 2-4) as the one closest to the definition of late drywell failures. Therefore,

$$CLASS\_7c\_DOSE = 4.55 \times 10^4 \text{ person-sv} * \frac{100 \text{ person-rem}}{1 \text{person-sv}} \quad \text{[Table 2-6]}$$

$$= 4.55 \times 10^6 \text{ person-rem}$$

Population Dose for EPRI Class 7d. The 50-miles population dose for the EPRI accident Class 7d (Severe Accident Phenomena Induced Late Torus Failures) is based on the Pilgrim Station collapsed accident progression bin 13 (Table 2-4) as the one closest to the definition of late torus failures. Therefore,

$$CLASS\_7d\_DOSE = 7.35 \times 10^3 \text{ person-sv} * \frac{100 \text{ person-rem}}{1 \text{person-sv}} \quad \text{[Table 2-6]}$$

$$= 7.35 \times 10^5 \text{ person-rem}$$

Population Dose for EPRI Class 8.

The 50-miles population dose for the EPRI accident Class 8 (Bypass) is based on the Pilgrim Station collapsed accident progression bin 19 (Table 2-4) as the one closest to the definition of bypass failure. This selection is based on the highest dose of all the containment failure collapsed accident progression bins, indicative of containment bypass scenarios. Therefore,

$$CLASS\_8\_DOSE = 5.66 \times 10^4 \text{ person-sv} * \frac{100 \text{ person-rem}}{1 \text{person-sv}} \quad \text{[Table 2-6]}$$

$$= 5.66 \times 10^6 \text{ person-rem}$$

Using the preceding information, the population dose for the 50-mile radius surrounding the Pilgrim Station is summarized in Table 2-7. (Note: the use of dose results for the 50-mile radius around the plant as a 'figure of merit' in the risk evaluation is consistent with past ILRT frequency extension submittals, and the NEI Interim Guidance [4]).

2.4.4 Baseline Population Dose Rate Estimate (Step 4)

This step calculates the baseline does rates for each of the eight EPRI's accident classes. The calculation is performed by multiplying the dose calculated in Step 3 (Table 2-7) by the associated frequency calculated in Step 1 (Table 2-5). Since the conditional containment pre-existing leakage probabilities for EPRI accident classes' 3a and 3b are based on a 3-per-10 year ILRT frequency, the calculated baseline results reflect a 3-per-10 year ILRT surveillance frequency.

$$CLASS\_1\_DOSE_{RATE} = CLASS\_1\_DOSE * CLASS\_1\_FREQUENCY$$
$$CLASS\_2\_DOSE_{RATE} = CLASS\_2\_DOSE * CLASS\_2\_FREQUENCY$$
$$CLASS\_3a\_DOSE_{RATE} = CLASS\_3a\_DOSE * CLASS\_3a\_FREQUENCY$$
$$CLASS\_3b\_DOSE_{RATE} = CLASS\_3b\_DOSE * CLASS\_3b\_FREQUENCY$$
$$CLASS\_7a\_DOSE_{RATE} = CLASS\_7a\_DOSE * CLASS\_7a\_FREQUENCY$$
$$CLASS\_7b\_DOSE_{RATE} = CLASS\_7b\_DOSE * CLASS\_7b\_FREQUENCY$$

$CLASS\_7c\_DOSE_{RATE}$ = CLASS_7c_DOSE * CLASS_7c_FREQUENCY
$CLASS\_7d\_DOSE_{RATE}$ = CLASS_7d_DOSE * CLASS_7d_FREQUENCY
$CLASS\_8\_DOSE_{RATE}$ = CLASS_8_DOSE * CLASS_8_FREQUENCY

Where:

$CLASS\_1\_DOSE_{RATE}$ = EPRI accident Class 1 dose rate given a 3-in-10 years ILRT interval
$CLASS\_2\_DOSE_{RATE}$ = EPRI accident Class 2 dose rate given a 3-in-10 years ILRT interval
$CLASS\_3a\_DOSE_{RATE}$ = EPRI accident Class 3a dose rate given a 3-in-10 years ILRT interval
$CLASS\_3b\_DOSE_{RATE}$ = EPRI accident Class 3b dose rate given a 3-in-10 years ILRT interval
$CLASS\_7a\_DOSE_{RATE}$ = EPRI accident Class 7a dose rate given a 3-in-10 years ILRT interval
$CLASS\_7b\_DOSE_{RATE}$ = EPRI accident Class 7b dose rate given a 3-in-10 years ILRT interval
$CLASS\_7c\_DOSE_{RATE}$ = EPRI accident Class 7c dose rate given a 3-in-10 years ILRT interval
$CLASS\_7d\_DOSE_{RATE}$ = EPRI accident Class 7d dose rate given a 3-in-10 years ILRT interval
$CLASS\_8\_DOSE_{RATE}$ = EPRI accident Class 8 dose rate given a 3-in-10 years ILRT interval

CLASS_1_DOSE = EPRI accident Class 1 dose = 1.06×10^{4} (person-rem) [Table 2-7]
CLASS_2_DOSE = EPRI accident Class 2 dose = 4.53×10^{6} (person-rem) [Table 2-7]
CLASS_3a_DOSE = EPRI accident Class 3a dose = 1.06×10^{5} (person-rem) [Table 2-7]
CLASS_3b_DOSE = EPRI accident Class 3b dose = 3.71×10^{5} (person-rem) [Table 2-7]
CLASS_7a_DOSE = EPRI accident Class 7a dose = 4.53×10^{6} (person-rem) [Table 2-7]
CLASS_7b_DOSE = EPRI accident Class 7b dose = 1.82×10^{6} (person-rem) [Table 2-7]
CLASS_7c_DOSE = EPRI accident Class 7c dose = 4.55×10^{6} (person-rem) [Table 2-7]
CLASS_7d_DOSE = EPRI accident Class 7d dose = 7.35×10^{5} (person-rem) [Table 2-7]
CLASS_8_DOSE = EPRI accident Class 8 dose = 5.66×10^{6} (person-rem) [Table 2-7]

CLASS_1_FREQUENCY = frequency of EPRI accident Class 1 given a 3-in-10 years ILRT interval
= 9.81×10^{-8}/ry [Table 2-5]

CLASS_2_FREQUENCY = frequency of EPRI accident Class 2 given a 3-in-10 years ILRT interval
= 4.42×10^{-11}/ry [Table 2-5]

CLASS_3a_FREQUENCY = frequency of EPRI accident Class 3a given a 3-in-10 years ILRT interval
= 1.18×10^{-8}/ry [Table 2-5]

CLASS_3b_FREQUENCY = frequency of EPRI accident Class 3b given a 3-in-10 years ILRT interval
= 1.18×10^{-9}/ry [Table 2-5]

CLASS_7a_FREQUENCY = frequency of EPRI accident Class 7a given a 3-in-10 years ILRT interval
= 1.59×10^{-7}/ry [Table 2-5]

CLASS_7b_FREQUENCY = frequency of EPRI accident Class 7b given a 3-in-10 years ILRT interval
= 2.19×10^{-8}/ry [Table 2-5]

CLASS_7c_FREQUENCY = frequency of EPRI accident Class 7c given a 3-in-10 years ILRT interval
= 4.38×10^{-6}/ry [Table 2-5]

CLASS_7d_FREQUENCY = frequency of EPRI accident Class 7d given a 3-in-10 years ILRT interval
= 1.70×10^{-6}/ry [Table 2-5]

CLASS_8_FREQUENCY = frequency of EPRI accident Class 8 given a 3-in-10 years ILRT interval
= 3.79×10^{-8}/ry [Table 2-5]

Therefore,

$CLASS\_1\_DOSE_{RATE}$	=	1.06×10^{4}	*	9.81×10^{-8}	=	1.04×10^{-3}	(person-rem/ry)
$CLASS\_2\_DOSE_{RATE}$	=	4.53×10^{6}	*	4.42×10^{-11}	=	2.00×10^{-4}	(person-rem/ry)
$CLASS\_3a\_DOSE_{RATE}$	=	1.06×10^{5}	*	1.18×10^{-8}	=	1.25×10^{-3}	(person-rem/ry)
$CLASS\_3b\_DOSE_{RATE}$	=	3.71×10^{5}	*	1.18×10^{-9}	=	4.38×10^{-4}	(person-rem/ry)
$CLASS\_7a\_DOSE_{RATE}$	=	4.53×10^{6}	*	1.59×10^{-7}	=	7.20×10^{-1}	(person-rem/ry)
$CLASS\_7b\_DOSE_{RATE}$	=	1.82×10^{6}	*	2.19×10^{-8}	=	3.99×10^{-2}	(person-rem/ry)
$CLASS\_7c\_DOSE_{RATE}$	=	4.55×10^{6}	*	4.38×10^{-6}	=	1.99×10^{1}	(person-rem/ry)
$CLASS\_7e\_DOSE_{RATE}$	=	7.35×10^{5}	*	1.70×10^{-6}	=	1.25×10^{0}	(person-rem/ry)
$CLASS\_8\_DOSE_{RATE}$	=	5.66×10^{6}	*	3.79×10^{-8}	=	2.15×10^{-1}	(person-rem/ry)

Table 2-8 summarizes the resulting baseline population dose rates by EPRI accident class.

2.4.5 Change in Probability of Detectable Leakage (Step 5)

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

According to NUREG-1493 [2] and per the NEI Interim Guidance [4], the calculation of the change in the probability of a pre-existing ILRT-detectable containment leakage is based on the relationship that relaxation of the ILRT interval results in increasing the average time that a pre-existing leak would exist undetected. Specifically, the relaxation of the Type A ILRT interval from 3-in-10 years to 1-in-10 years will increase the average time that a leak detectable only by an ILRT goes undetected from 18 to 60 months[6], a factor of 3.333 increase (60/18). Therefore, the change in probability of leakage due to the ILRT interval extension is calculated by applying a multiplier factor determined by the ratio of the average times of undetection for the two ILRT interval cases.

From Section 2.3 "Input and Design Criteria", the calculated pre-existing ILRT detectable leakage probabilities based on 3 in-10 years ILRT frequency is 0.027 for small pre-existing leakage (EPRI accident class 3a) and 0.0027 for large pre-existing leakage (EPRI accident class 3b).

Since October 1996, the Pilgrim Station plant has been operating under a 1-in-10 years ILRT testing frequency consistent with the performance-based Option B of 10 CFR Part 50, Appendix J. [13]. As a result, the baseline leakage probabilities, (which are based on a 3-in-10 years ILRT frequency) must be revised to reflect the current 1-in-10 years Pilgrim ILRT testing frequency. This is performed as follows:

$$PROB_{class\_3a\_10} = PROB_{class\_3a} * \left[\frac{SURTEST_{10}}{18}\right]$$

$$PROB_{class\_3b\_10} = PROB_{class\_3b} * \left[\frac{SURTEST_{10}}{18}\right]$$

[6] Multiplying the test interval by ½ and multiplying by 12 to convert from a year to months calculates the average time for undetection.

Where:

$PROB_{class\_3a\_10}$ = probability of small pre-existing containment liner leakage given a 1-in-10 years ILRT frequency.

$PROB_{class\_3a}$ = probability of small pre-existing containment liner leakage given a 3-in-10 years ILRT frequency = 0.027 [Section 2.3, input#8]

$PROB_{class\_3b}$ = probability of large pre-existing containment liner leakage given a 3-in-10 years ILRT frequency = 0.0027 [Section 2.3, input #9]

$SURTEST_{10}$ = surveillance interval of interest, months/2 = 10 years*12 $\frac{\text{months}}{\text{year}}$/2 = 60 months

Therefore,

$$PROB_{class\_3a\_10} = 0.027 * \left[\frac{60}{18}\right] = 0.09$$

$$PROB_{class\_3b\_10} = 0.0027 * \left[\frac{60}{18}\right] = 0.009$$

Similarly, the pre-existing ILRT detectable leakage probabilities for the 1-in-15 years ILRT frequency being analyzed by Pilgrim are calculated as follows:

$$PROB_{class\_3a\_15} = PROB_{class\_3a} * \frac{SURTEST_{15}}{18}$$

$$PROB_{class\_3b\_15} = PROB_{class\_3b} * \left[\frac{SURTEST_{15}}{18}\right]$$

Where:

$PROB_{class\_3a\_15}$ = probability of small pre-existing containment liner leakage given a 1-in-15 years ILRT frequency.

$PROB_{class\_3a}$ = probability of small pre-existing containment liner leakage given a 3-in-10 years ILRT frequency = 0.027 [Section 2.3, input#6]

$PROB_{class\_3b}$ = probability of large pre-existing containment liner leakage given a 3-in-10 years ILRT frequency = 0.0027 [Section 2.3, input #7]

$SURTEST_{15}$ = surveillance interval of interest, months/2 = 15 years*12 $\frac{\text{months}}{\text{year}}$/2 = 90 months

Therefore,

$$PROB_{class\_3a\_15} = 0.027 * \left[\frac{90}{18}\right] = 0.135$$

$$PROB_{class\_3b\_15} = 0.0027 * \left[\frac{90}{18}\right] = 0.0135$$

Given the above revised leakage probabilities, the frequencies of the EPRI accident classes calculated in Step 1, also needs to be revised to reflect the increase change in leakage probabilities.

As previously stated, Type A tests impact only Class 1 and Class 3 sequences. Therefore, EPRI accident Class 1 frequency changes are calculated similar to Step 1, and the rest of EPRI's Classes; 2, 7 and 8 remain the same.

Revised Frequency of EPRI Class 3a Sequences. Consistent with NEI Interim Guidance [4], the frequency per reactor year for this category is calculated as:

$$CLASS\_3a\_FREQUENCY_{10} = PROB_{class\_3a\_10} * [CDF - (CDF_{LERF} + CDF_{NO\_LERF})]$$

$$CLASS\_3a\_FREQUENCY_{15} = PROB_{class\_3a\_15} * [CDF - (CDF_{LERF} + CDF_{NO\_LERF})]$$

Where:

$CLASS\_3a\_FREQUENCY_{\_10}$ = frequency of small pre-existing containment liner leakage given a 1-in-10 years ILRT interval

$CLASS\_3a\_FREQUENCY_{\_15}$ = frequency of small pre-existing containment liner leakage given a 1-in-15 years ILRT interval

$PROB_{class\_3a\_10}$ = probability of small pre-existing containment liner leakage given a 1-in-10 years ILRT frequency = 0.09 [See above write-up]

$PROB_{class\_3a\_15}$ = probability of small pre-existing containment liner leakage given a 1-in-15 years ILRT frequency = 0.135 [See above write-up]

CDF = Pilgrim Station PSA point estimate core damage frequency
= 6.41×10^{-6}/ry [Section 2.3, Input#2]

CDF_{LERF} = CDF for those individual sequences that independently cause a LERF.
= 1.13×10^{-7}/ry (See step 1 write-up)

CDF_{NO_LERF} = CDF for those individual sequences that never cause a LERF. This is denoted from the loss of containment heat removal accident sequences (Pilgrim Station Class II)
= 5.86×10^{-6}/ry [Table 2-1]

Therefore,

$$CLASS\_3a\_FREQUENCY_{10} = 0.09 * [6.41 \times 10^{-6}/ry - (1.13 \times 10^{-7}/ry + 5.86 \times 10^{-6}/ry)] = 3.93 \times 10^{-8}/ry$$

$$CLASS\_3a\_FREQUENCY_{15} = 0.135 * [6.41 \times 10^{-6}/ry - (1.13 \times 10^{-7}/ry + 5.86 \times 10^{-6}/ry)] = 5.90 \times 10^{-8}/ry$$

Frequency of EPRI Class 3b Sequences. Consistent with NEI Interim Guidance [4], the frequency per reactor year for this category is calculated as:

$$CLASS\_3b\_FREQUENCY_{10} = PROB_{class\_3b\_10} * CDF$$

$$CLASS\_3b\_FREQUENCY_{15} = PROB_{class\_3b\_15} * CDF$$

Where:

$CLASS\_3b\_FREQUENCY_{\_10}$ = frequency of small pre-existing containment liner leakage given a 1-in-10 years ILRT interval

$CLASS\_3b\_FREQUENCY_{\_15}$ = frequency of small pre-existing containment liner leakage given a 1-in-15 years ILRT interval

$PROB_{class\_3b\_10}$ = probability of small pre-existing containment liner leakage given a 1-in-10 years ILRT frequency = 0.009 [See above write-up]

$PROB_{class\_3b\_10}$ = probability of small pre-existing containment liner leakage given a 1-in-15 years ILRT frequency = 0.0135 [See above write-up]

CDF = Pilgrim IPE core damage frequency = 6.41×10^{-6}/ry [Section 2.3, input # 2]

CDF_{LERF} = CDF for those individual sequences that independently cause a LERF.
= 1.13×10^{-7}/ry (See step 1 write-up)

CDF_{NO_LERF} = CDF for those individual sequences that never cause a LERF. This is denoted from the loss of containment heat removal accident sequences (Pilgrim Station Class II)
= 5.86×10^{-6}/ry [Table 2-1]

Therefore,

$$CLASS\_3b\_FREQUENCY_{10} = 0.009 * [6.41 \times 10^{-6}/ry - (1.13 \times 10^{-7}/ry + 5.86 \times 10^{-6}/ry)] = 3.93 \times 10^{-9}/ry$$

$$CLASS\_3b\_FREQUENCY_{15} = 0.0135 * [6.41 \times 10^{-6}/ry - (1.13 \times 10^{-7}/ry + 5.86 \times 10^{-6}/ry)] = 5.90 \times 10^{-9}/ry$$

Frequency of EPRI Class 1 Sequences. Consistent with NEI Interim Guidance [4], the frequency per reactor year for these sequences is calculated by subtracting the frequencies of EPRI Categories 3a and 3b from the sum of all severe accident progression sequence frequencies in which the containment is isolated and intact:

$$CLASS\_1\_FREQUENCY_{10} = NCF - CLASS\_3a\_FREQUENCY_{10} - CLASS\_3b\_FREQUENCY_{10}$$

$$CLASS\_1\_FREQUENCY_{15} = NCF - CLASS\_3a\_FREQUENCY_{15} - CLASS\_3b\_FREQUENCY_{15}$$

Where:

NCF = frequency in which containment leakage is at or below maximum allowable Technical Specification Leakage = 1.11×10^{-7} /ry [Table 2-2]

$CLASS\_1\_FREQUENCY_{10}$ = frequency of no containment failure given a 1-in-10 years ILRT interval

$CLASS\_1\_FREQUENCY_{15}$ = frequency of no containment failure given a 1-in-15 years ILRT interval

$CLASS\_3a\_FREQUENCY_{10}$ = frequency of small pre-existing containment liner leakage given a 1-in-10 years ILRT interval = 3.93×10^{-8}/ry [See above write-up]

$CLASS\_3b\_FREQUENCY_{10}$ = frequency of small pre-existing containment liner leakage given a 1-in-10 years ILRT interval = 3.93×10^{-9}/ry [See above write-up]

$CLASS\_3a\_FREQUENCY_{15}$ = frequency of small pre-existing containment liner leakage given a 1-in-10 years ILRT interval = 5.90×10^{-8}/ry [See above write-up]

$CLASS\_3b\_FREQUENCY_{15}$ = frequency of small pre-existing containment liner leakage given a 1-in-10 years ILRT interval = 5.90×10^{-9}/ry [See above write-up]

Therefore:

$$CLASS\_1\_FREQUENCY_{10} = 1.11 \times 10^{-7} - 3.93 \times 10^{-8}/ry - 3.93 \times 10^{-9}/ry = 6.78 \times 10^{-8}/ry$$

$$CLASS\_1\_FREQUENCY_{15} = 1.11 \times 10^{-7} - 5.90 \times 10^{-8}/ry - 5.90 \times 10^{-9}/ry = 4.61 \times 10^{-8}/ry$$

The impacted frequencies of the EPRI accident classes are summarized in Table 2-9.

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

This step, per the NEI Interim Guidance [4], calculates the population dose rate for the new surveillance intervals of interest by multiplying the population dose (Table 2-7) by the frequency for each of the eight EPRI's accident classes (Tables 2-5 and 2-9). In addition, sum the accident class dose rates to obtain the total dose rate.

Per the NEI Interim Guidance [4], EPRI accident Classes 4 (Small Isolation Failure - failure-to-seal, Type B test), 5 (Small Isolation Failure - failure-to-seal, Type C test), and 6 (Containment Isolation Failures, dependent failures, personnel errors) are not affected by ILRT frequency and are not analyzed as part of this risk assessment. Therefore no selections of population dose estimates are made for these accident classes.

The calculation for a 1-in-10 years ILRT interval is as follows:

$$CLASS\_1\_DOSE_{RATE\text{-}10} = CLASS\_1\_DOSE * CLASS\_1\_FREQUENCY_{10}$$
$$CLASS\_2\_DOSE_{RATE\text{-}10} = CLASS\_2\_DOSE * CLASS\_2\_FREQUENCY_{10}$$
$$CLASS\_3a\_DOSE_{RATE\text{-}10} = CLASS\_3a\_DOSE * CLASS\_3a\_FREQUENCY_{10}$$
$$CLASS\_3b\_DOSE_{RATE\text{-}10} = CLASS\_3b\_DOSE * CLASS\_3b\_FREQUENCY_{10}$$
$$CLASS\_7a\_DOSE_{RATE\text{-}10} = CLASS\_7a\_DOSE * CLASS\_7a\_FREQUENCY_{10}$$
$$CLASS\_7b\_DOSE_{RATE\text{-}10} = CLASS\_7b\_DOSE * CLASS\_7b\_FREQUENCY_{10}$$
$$CLASS\_7c\_DOSE_{RATE\text{-}10} = CLASS\_7c\_DOSE * CLASS\_7c\_FREQUENCY_{10}$$
$$CLASS\_7d\_DOSE_{RATE\text{-}10} = CLASS\_7d\_DOSE * CLASS\_7d\_FREQUENCY_{10}$$
$$CLASS\_8\_DOSE_{RATE\text{-}10} = CLASS\_8\_DOSE * CLASS\_8\_FREQUENCY_{10}$$

Where:

$CLASS\_1\_DOSE_{RATE\text{-}10}$ = EPRI accident Class 1 dose rate given a 1-in-10 years ILRT interval
$CLASS\_2\_DOSE_{RATE\text{-}10}$ = EPRI accident Class 2 dose rate given a 1-in-10 years ILRT interval

$CLASS\_3a\_DOSE_{RATE\text{-}10}$ = EPRI accident Class 3a dose rate given a 1-in-10 years ILRT interval
$CLASS\_3b\_DOSE_{RATE\text{-}10}$ = EPRI accident Class 3b dose rate given a 1-in-10 years ILRT interval
$CLASS\_7a\_DOSE_{RATE\text{-}10}$ = EPRI accident Class 7a dose rate given a 1-in-10 years ILRT interval
$CLASS\_7b\_DOSE_{RATE\text{-}10}$ = EPRI accident Class 7b dose rate given a 1-in-10 years ILRT interval
$CLASS\_7c\_DOSE_{RATE\text{-}10}$ = EPRI accident Class 7c dose rate given a 1-in-10 years ILRT interval
$CLASS\_7d\_DOSE_{RATE\text{-}10}$ = EPRI accident Class 7d dose rate given a 1-in-10 years ILRT interval
$CLASS\_8\_DOSE_{RATE\text{-}10}$ = EPRI accident Class 8 dose rate given a 1-in-10 years ILRT interval

CLASS_1_DOSE = EPRI accident Class 1 dose = 1.06×10^{4} (person-rem) [Table 2-7]
CLASS_2_DOSE = EPRI accident Class 2 dose = 4.53×10^{6} (person-rem) [Table 2-7]
CLASS_3a_DOSE = EPRI accident Class 3a dose = 1.06×10^{5} (person-rem) [Table 2-7]
CLASS_3b_DOSE = EPRI accident Class 3b dose = 3.71×10^{5} (person-rem) [Table 2-7]
CLASS_7a_DOSE = EPRI accident Class 7a dose = 4.53×10^{6} (person-rem) [Table 2-7]
CLASS_7b_DOSE = EPRI accident Class 7b dose = 1.82×10^{6} (person-rem) [Table 2-7]
CLASS_7c_DOSE = EPRI accident Class 7c dose = 4.55×10^{6} (person-rem) [Table 2-7]
CLASS_7d_DOSE = EPRI accident Class 7d dose = 7.35×10^{5} (person-rem) [Table 2-7]
CLASS_8_DOSE = EPRI accident Class 8 dose = 5.66×10^{6} (person-rem) [Table 2-7]

$CLASS\_1\_FREQUENC\ Y_{10}$ = frequency of EPRI accident Class 1 given a 1-in-10 years ILRT Interval = 6.78×10^{-8}/ry [Table 2-9]

$CLASS\_2\_FREQUENC\ Y_{10}$ = frequency of EPRI accident Class 2 given a 3-in-10 years ILRT Interval = 4.42×10^{-11}/ry [Table 2-5]

$CLASS\_3a\_FREQUENC\ Y_{10}$ = frequency of EPRI accident Class 3a given a 1-in-10 years ILRT Interval = 3.93×10^{-8}/ry [Table 2-9]

$CLASS\_3b\_FREQUENC\ Y_{10}$ = frequency of EPRI accident Class 3b given a 1-in-10 years ILRT Interval = 3.93×10^{-9}/ry [Table 2-9]

$CLASS\_7a\_FREQUENC\ Y_{10}$ = frequency of EPRI accident Class 7a given a 3-in-10 years ILRT Interval = 1.59×10^{-7}/ry [Table 2-5]

$CLASS\_7b\_FREQUENC\ Y_{10}$ = frequency of EPRI accident Class 7b given a 3-in-10 years ILRT Interval = 2.19×10^{-8}/ry [Table 2-5]

$CLASS\_7c\_FREQUENC\ Y_{10}$ = frequency of EPRI accident Class 7c given a 3-in-10 years ILRT Interval = 4.38×10^{-6}/ry [Table 2-5]

$CLASS\_7d\_FREQUENC\ Y_{10}$ = frequency of EPRI accident Class 7d given a 3-in-10 years ILRT Interval = 1.70×10^{-6}/ry [Table 2-5]

$CLASS\_8\_FREQUENC\ Y_{10}$ = frequency of EPRI accident Class 8 given a 3-in-10 years ILRT Interval = 3.79×10^{-8}/ry [Table 2-5]

Therefore,

$CLASS\_1\_DOSE_{RATE\text{-}10} = 1.06 \times 10^{4} * 6.78 \times 10^{-8} = 7.20 \times 10^{-4}$ (person-rem/ry)
$CLASS\_2\_DOSE_{RATE\text{-}10} = 4.53 \times 10^{6} * 4.42 \times 10^{-11} = 2.00 \times 10^{-4}$ (person-rem/ry)
$CLASS\_3a\_DOSE_{RATE\text{-}10} = 1.06 \times 10^{5} * 3.93 \times 10^{-8} = 4.17 \times 10^{-3}$ (person-rem/ry)
$CLASS\_3b\_DOSE_{RATE\text{-}10} = 3.71 \times 10^{5} * 3.93 \times 10^{-9} = 1.46 \times 10^{-3}$ (person-rem/ry)
$CLASS\_7a\_DOSE_{RATE\text{-}10} = 4.53 \times 10^{6} * 1.59 \times 10^{-7} = 7.20 \times 10^{-1}$ (person-rem/ry)
$CLASS\_7b\_DOSE_{RATE\text{-}10} = 1.82 \times 10^{6} * 2.19 \times 10^{-8} = 3.99 \times 10^{-2}$ (person-rem/ry)

$CLASS\_7c\_DOSE_{RATE\text{-}10}$ = 4.55×10^{6} * 4.38×10^{-6} = 1.99×10^{1} (person-rem/ry)
$CLASS\_7d\_DOSE_{RATE\text{-}10}$ = 7.35×10^{5} * 1.70×10^{-6} = 1.25×10^{0} (person-rem/ry)
$CLASS\_8\_DOSE_{RATE\text{-}10}$ = 5.66×10^{6} * 3.79×10^{-8} = 2.15×10^{-1} (person-rem/ry)

The calculation for a 1-in-15 years ILRT interval is as follows for the:

$CLASS\_1\_DOSE_{RATE\text{-}15}$ = CLASS_1_DOSE * $CLASS\_1\_FREQUENCY_{15}$
$CLASS\_2\_DOSE_{RATE\text{-}15}$ = CLASS_2_DOSE * $CLASS\_2\_FREQUENCY_{15}$
$CLASS\_3a\_DOSE_{RATE\text{-}15}$ = CLASS_3a_DOSE * $CLASS\_3a\_FREQUENCY_{15}$
$CLASS\_3b\_DOSE_{RATE\text{-}15}$ = CLASS_3b_DOSE * $CLASS\_3b\_FREQUENCY_{15}$
$CLASS\_7a\_DOSE_{RATE\text{-}15}$ = CLASS_7a_DOSE * $CLASS\_7a\_FREQUENCY_{15}$
$CLASS\_7b\_DOSE_{RATE\text{-}15}$ = CLASS_7b_DOSE * $CLASS\_7b\_FREQUENCY_{15}$
$CLASS\_7c\_DOSE_{RATE\text{-}15}$ = CLASS_7c_DOSE * $CLASS\_7c\_FREQUENCY_{15}$
$CLASS\_7d\_DOSE_{RATE\text{-}15}$ = CLASS_7d_DOSE * $CLASS\_7d\_FREQUENCY_{15}$
$CLASS\_8\_DOSE_{RATE\text{-}15}$ = CLASS_8_DOSE * $CLASS\_8\_FREQUENCY_{15}$

Where:

CLASS_1_DOSE = EPRI accident Class 1 dose = 1.06×10^{4} (person-rem) [Table 2-7]
CLASS_2_DOSE = EPRI accident Class 2 dose = 4.53×10^{6} (person-rem) [Table 2-7]
CLASS_3a_DOSE = EPRI accident Class 3a dose = 1.06×10^{5} (person-rem) [Table 2-7]
CLASS_3b_DOSE = EPRI accident Class 3b dose = 3.71×10^{5} (person-rem) [Table 2-7]
CLASS_7a_DOSE = EPRI accident Class 7a dose = 4.53×10^{6} (person-rem) [Table 2-7]
CLASS_7b_DOSE = EPRI accident Class 7b dose = 1.82×10^{6} (person-rem) [Table 2-7]
CLASS_7c_DOSE = EPRI accident Class 7c dose = 4.55×10^{6} (person-rem) [Table 2-7]
CLASS_7d_DOSE = EPRI accident Class 7d dose = 7.35×10^{5} (person-rem) [Table 2-7]
CLASS_8_DOSE = EPRI accident Class 8 dose = 5.66×10^{6} (person-rem) [Table 2-7]

$CLASS\_1\_FREQUENCY_{15}$ = frequency of EPRI accident Class 1 given a 1-in-15 years ILRT Interval = 4.61×10^{-8}/ry [Table 2-9]

$CLASS\_2\_FREQUENCY_{15}$ = frequency of EPRI accident Class 2 given a 3-in-10 years ILRT Interval = 4.42×10^{-11}/ry [Table 2-5]

$CLASS\_3a\_FREQUENCY_{15}$ = frequency of EPRI accident Class 3a given a 1-in-15 years ILRT Interval = 5.90×10^{-8}/ry [Table 2-9]

$CLASS\_3b\_FREQUENCY_{15}$ = frequency of EPRI accident Class 3b given a 1-in-15 years ILRT Interval = 5.90×10^{-9}/ry [Table 2-9]

$CLASS\_7a\_FREQUENCY_{15}$ = frequency of EPRI accident Class 7a given a 3-in-10 years ILRT Interval = 1.59×10^{-7}/ry [Table 2-5]

$CLASS\_7b\_FREQUENCY_{15}$ = frequency of EPRI accident Class 7b given a 3-in-10 years ILRT Interval = 2.19×10^{-8}/ry [Table 2-5]

$CLASS\_7c\_FREQUENCY_{15}$ = frequency of EPRI accident Class 7c given a 3-in-10 years ILRT Interval = 4.38×10^{-6}/ry [Table 2-5]

$CLASS\_7d\_FREQUENCY_{15}$ = frequency of EPRI accident Class 7d given a 3-in-10 years ILRT Interval = 1.70×10^{-6}/ry [Table 2-5]

CLASS_8_FREQUENC Y_{15} = frequency of EPRI accident Class 8 given a 3-in-10 years ILRT Interval = 3.79×10^{-8}/ry [Table 2-5]

Therefore,

CLASS_1_$DOSE_{RATE\text{-}15}$	=	1.06×10^{4}	*	4.61×10^{-8}	=	4.89×10^{-4} (person-rem/ry)
CLASS_2_$DOSE_{RATE\text{-}15}$	=	4.53×10^{6}	*	4.42×10^{-11}	=	2.00×10^{-4} (person-rem/ry)
CLASS_3a_$DOSE_{RATE\text{-}15}$	=	1.06×10^{5}	*	5.90×10^{-8}	=	6.25×10^{-3} (person-rem/ry)
CLASS_3b_$DOSE_{RATE\text{-}15}$	=	3.71×10^{5}	*	5.90×10^{-9}	=	2.19×10^{-3} (person-rem/ry)
CLASS_7a_$DOSE_{RATE\text{-}15}$	=	4.53×10^{6}	*	1.59×10^{-7}	=	7.20×10^{-1} (person-rem/ry)
CLASS_7b_$DOSE_{RATE\text{-}15}$	=	1.82×10^{6}	*	2.19×10^{-8}	=	3.99×10^{-2} (person-rem/ry)
CLASS_7c_$DOSE_{RATE\text{-}15}$	=	4.55×10^{6}	*	4.38×10^{-6}	=	1.99×10^{1} (person-rem/ry)
CLASS_7d_$DOSE_{RATE\text{-}15}$	=	7.35×10^{5}	*	1.70×10^{-6}	=	1.25×10^{0} (person-rem/ry)
CLASS_8_$DOSE_{RATE\text{-}15}$	=	5.66×10^{6}	*	3.79×10^{-8}	=	2.15×10^{-1} (person-rem/ry)

The dose rates per EPRI accident class as a function of ILRT interval are summarized in Table 2-10.

2.4.7 Change In Population Dose Rate Due to New ILRT Interval (Step 7)

This step, per the NEI Interim Guidance [4] calculates the percentage of the total dose rate attributable to EPRI accident Classes 3a and 3b (those accident classes affected by change in ILRT surveillance interval) and the change in this result dose rate from the base dose rate attributable to changes in ILRT surveillance interval.

Based on the results summarized in Table 2-10, for the current Pilgrim Station 1-in10 years ILRT interval, the percentage contribution to total dose rate from EPRI's accident Classes 3a and 3b is calculated as follows:

PER_CHG_{10} = percentage contribution to total dose rate from EPRI's accident Classes 3a and 3b given a 1-in-10 years ILRT interval

CLASS_3a_$DOSE_{RATE\text{-}10}$ = EPRI accident Class 3a dose rate given a 1-in-10 years ILRT interval
= 4.17×10^{-3} [Table 10]

CLASS_3b_$DOSE_{RATE\text{-}10}$ = EPRI accident Class 3b dose rate given a 1-in-10 years ILRT interval
= 1.46×10^{-3} [Table 10]

TOT- $DOSE_{RATE\text{-}10}$ = Total dose rate for all EPRI's Classes given a 1-in-10 years ILRT interval
= 22.132 [Table 10]

Therefore,

$$PER\_CHG_{10} = \left[\frac{4.17 \times 10^{-3} + 1.46 \times 10^{-3}}{22.132} \right] * 100$$

$$PER\_CHG_{10} = 0.0254\%$$

The percentage contribution to total dose rate from EPRI's accident Classes 3a and 3b based on the propose 1-in-15 years ILRT interval is calculated as follows:

$$PER\_CHG_{15} = \left[\frac{CLASS\_3a\_DOSE_{RATE\text{-}15} + CLASS\_3b\_DOSE_{RATE\text{-}15}}{TOT\text{-}DOSE_{RATE\text{-}15}} \right] * 100$$

Where:

$CLASS\_3a\_DOSE_{RATE\text{-}15}$ = EPRI accident Class 3a dose rate given a 1-in-15 years ILRT interval
= 6.25×10^{-3} (person-rem/ry) [Table 2-10]

$CLASS\_3b\_DOSE_{RATE\text{-}15}$ = EPRI accident Class 3b dose rate given a 1-in-15 years ILRT interval
= 2.19×10^{-3} (person-rem/ry) [Table 2-10]

$TOT\text{-}DOSE_{RATE\text{-}15}$ = Total dose rate for all EPRI's Classes given a 1-in-15 years ILRT interval
= 22.134 (person-rem/ry) [Table 2-10]

Therefore,

$$PER\_CHG_{15} = \left[\frac{6.25 \times 10^{-3} + 2.19 \times 10^{-3}}{22.134} \right] * 100$$

$$PER\_CHG_{15} = 0.038\%$$

Based on the above results, the changes from the 1-in-10 years to 1-in-15 years dose rate is as follows:

$$INCREASE_{10\text{-}15} = \left[\frac{TOT\text{-}DOSE_{RATE\text{-}15} - TOT\text{-}DOSE_{RATE\text{-}10}}{TOT\text{-}DOSE_{RATE\text{-}10}} \right] * 100$$

Where:

$INCREASE_{10\text{-}15}$ = percent change from 1-in-10 years ILRT interval to 1-in-15 years ILRT interval

$TOT\text{-}DOSE_{RATE\text{-}15}$ = Total dose rate for all EPRI's Classes given a 1-in-10 years ILRT interval
= 22.134 (person-rem/ry) [Table 2-10]

$TOT\text{-}DOSE_{RATE\text{-}10}$ = Total dose rate for all EPRI's Classes given a 1-in-10 years ILRT interval
= 22.132 (person-rem/ry) [Table 2-10]

Therefore,

$$INCREASE_{10\text{-}15} = \left[\frac{22.134 - 22.132}{22.132} \right] * 100 = 0.009\%$$

The above increase in risk on the total integrated plant risk for those accident sequences influenced by Type A testing, given the change from a 1-in-10 years test interval to a 1-in-15 years test interval, is found to be 0.009%. This value can be considered to be a negligible increase in risk.

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

This step, per the NEI Interim Guidance [4] calculates the change in the large early release frequency with extending the ILRT interval from 1-in-10 years to 1-in15-years.

The risk impact 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 containment could in fact result in a large release due to failure to detect a pre-existing leak during the relaxation period. For this evaluation only accident Class 3 sequences have the potential to result in large releases if a pre-existing leak were present. Class 1 sequences are not considered as potential large release pathways because for these sequences the containment remains intact. Therefore, the containment leak rate is expected to be small (less than 2La). A larger leak rate would imply an impaired containment, such as classes 2, 3, 6 and 7.

Late releases are excluded regardless of the size of the leak because late releases are, by definition, not a LERF event. At the same time, sequences in the Pilgrim PSA [7], which result in large releases (e.g., large isolation valve failures), are not impacted because a LERF will occur regardless of the presence of a pre-existing leak. Therefore, the frequency of accident Class 3b sequences (Table 2-9) is used as the LERF for Pilgrim.

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

$$)LERF_{10\text{-}15} = CLASS\_3b\_FREQUENC\ Y_{15} - CLASS\_3b\_FREQUENC\ Y_{10}$$

Where:

$)LERF_{10\text{-}15}$ = the change in LERF from 1-in-10 years ILRT interval to 1-in-15 years ILRT interval

$CLASS\_3b\_FREQUENC\ Y_{15}$ = frequency of EPRI accident Class 3b given a 1-in-15 years ILRT Interval = 5.90×10^{-9} /ry [Table 2-9]

$CLASS\_3b\_FREQUENC\ Y_{10}$ = frequency of EPRI accident Class 3b given a 1-in-10 years ILRT Interval = 3.93×10^{-9}/ry [Table 2-9]

Therefore,

$$)LERF_{10\text{-}15} = 5.90 \times 10^{-9} - 3.93 \times 10^{-9}$$

$$)LERF_{10\text{-}15} = 1.97 \times 10^{-9}/ry$$

Regulatory Guide 1.174 [6] provides guidance for determining the risk impact of plant-specific changes to the licensing basis. Regulatory Guide 1.174 [5] 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. Since the ILRT does not impact CDF, the relevant risk metric is LERF.

This)LERF of 1.97×10^{-8}/ry falls into Region III, Very Small Change in Risk (Figure 2-1), of the acceptance guidelines in NRC Regulatory Guide 1.174 [6]. Therefore, because Regulatory Guide 1.174 [6] defines very small changes in LERF as below 10^{-7}/yr, increasing the ILRT interval at Pilgrim from the currently allowed 1-in-10 years to 1-in-15 years is non-risk significant from a risk perspective.

It should be noted that if the risk increase is measured from the original 3-in-10-year interval, the increase in LERF is as follows:

$)LERF_{3\text{-}15}$ = CLASS_3b_FREQUENC Y_{15} - CLASS_3b_FREQUENC Y_3

Where:

$)LERF_{3\text{-}15}$ = the change in LERF from 3-in-10 years ILRT interval to 1-in-15 years ILRT interval

CLASS_3b_FREQUENC Y_{15} = frequency of EPRI accident Class 3b given a 1-in-15 years ILRT Interval = 5.90×10^{-9}/ry [Table 2-9]

CLASS_3b_FREQUENC Y_3 = frequency of EPRI accident Class 3b given a 1-in-10 years ILRT Interval = 1.18×10^{-9}/ry [Table 2-9]

Therefore,

$$)LERF_{3\text{-}15} = 5.90 \times 10^{-9} - 1.18 \times 10^{-9}$$

$$)LERF_{3\text{-}15} = 4.72 \times 10^{-9}/ry$$

Similar to the $)LERF_{10\text{-}15}$ result, the $)LERF_{3\text{-}15}$ is also non-risk significant from a risk perspective.

2.4.9 Impact on Conditional Containment Failure Probability (Step 9)

This step, per the NEI Interim Guidance [4] calculates the change in conditional containment failure probability (CCFP). The CCFP risk metric ensures and shows that the proposed change in ILRT interval is consistent with the defense-in-depth philosophy describe in Regulatory Guide 1.174 [6][7].

In this calculation, the change in CCFP tracts the impact of the ILRT on both early (LERF) and late radionuclide releases. Based on the NEI Interim Guidance [4], CCFP consists of all those accident sequences resulting in a radionuclide release other that the intact containment state for EPRI accident Class 1, and small failures state for EPRI accident Class 3a. In addition, the CCFP is conditional given a severe core damage accident. The change in CCFP is calculated by the following equation:

CCFP=1-(Intact Containment Frequency/Total CDF)

Or

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

For the 1-in-10 years ILRT interval:

$$CCFP_{10} = \left\{ 1 - \left(\left[\frac{CLASS\_1\_FREQUENC\ Y_{10} + CLASS\_3a\_FREQUENCY_{10}}{CDF} \right] \right) \right\} * 100\%$$

Where:

$CCFP_{10}$ = conditional containment failure probability given 1-in-10 years ILRT interval

CDF = Pilgrim Station PSA point estimate core damage frequency = 6.41×10^{-6}/ry [Section 2.3, input#2]

CLASS_1_FREQUENC Y_{10} = frequency of EPRI accident Class 1 given a 1-in-10 years ILRT Interval = 6.78×10^{-8}/ry [Table 2-9]

[7] The defense-in-depth philosophy is maintained as a reasonable balance among prevention of core damage, containment failure and consequence mitigation.

CLASS_3a_FREQUENC Y_{10} = frequency of EPRI accident Class 3a given a 1-in-10 years ILRT Interval = 3.93×10^{-8}/ry [Table 2-9]

Therefore,

$$CCFP_{10} = \left\{ 1 - \left[\frac{6.78 \times 10^{-8} + 3.93 \times 10^{-8}}{6.41 \times 10^{-6}} \right] \right\} * 100\%$$

$$CCFP_{10} = 98.33\%$$

For the 1-in-15 years ILRT interval:

$$CCFP_{15} = \left\{ 1 - \left(\left[\frac{CLASS\_1\_FREQUENC\ Y_{15} + CLASS\_3a\_FREQUENCY_{15}}{CDF} \right] \right) \right\} * 100\%$$

Where:

$CCFP_{15}$ = conditional containment failure probability given 1-in-15 years ILRT interval

CDF = Pilgrim Station PSA point estimate core damage frequency = 6.41×10^{-6}/ry [Section 5, input#2]

CLASS_1_FREQUENC Y_{15} = frequency of EPRI accident Class 1 given a 1-in-15 years ILRT Interval = 4.61×10^{-8}/ry [Table 2-9]

CLASS_3a_FREQUENC Y_{15} = frequency of EPRI accident Class 3a given a 1-in-15 years ILRT Interval = 5.90×10^{8}/ry [Table 2-9]

Therefore,

$$CCFP_{15} = \left\{ 1 - \left[\frac{4.61 \times 10^{-8} + 5.90 \times 10^{-8}}{6.41 \times 10^{-6}} \right] \right\} * 100\%$$

$$CCFP_{15} = 98.36\%$$

Therefore, the change in the conditional containment failure probability from 1-in-10 years to 1-in-15 years is:

$$)CCFP_{10\text{-}15} = CCFP_{15} - CCFP_{10}$$

$$)CCFP_{10\text{-}15} = 98.36\% - 98.33\%$$

$$)CCFP_{10\text{-}15} = 0.03\%$$

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

2.5 External Events Impact

In response to Generic Letter 88-20, Supplement 4 [14], Pilgrim submitted an Individual Plant Examination of External Events (IPEEE) in July 1994 [15]. The IPEEE was a review of external hazard risk (i.e., seismic, fires, high winds, external flooding, etc) to identify potential plant vulnerabilities and to understand severe accident risks. The results of the Pilgrim Station IPEEE are therefore used in this risk assessment to provide a comparison of the effect of external hazards when extending the current 1-in-10 years to 1-in-15 years Type A ILRT interval.

The Pilgrim Station IPEEE submittal [15] examined a spectrum of external events hazards based on acceptable screening methods (Seismic PRA [16, 17], EPRI Fire PRA methodology [19], etc.). These screening methods use varying levels of conservatism; therefore, it is not practical to incorporate realistic quantitative risk assessments of all external event hazards into the ILRT extension assessment at this time. As a result, external events hazards are evaluated as a sensitivity case to demonstrate that the conclusions of the internal events analysis would not be changed if external events hazards were considered.

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

The purpose of the external events evaluation is to determine whether there are any unique insights or important quantitative information that explicitly impact the risk assessment results when considering only internal events.

The quantitative consideration of external hazards is discussed in more detail in Appendix A of this report. As can be seen from Appendix A, if the external hazard risk results of the Pilgrim Station IPEEE are included in this assessment (i.e., in addition to internal events), the change in LERF associated with the increase in ILRT interval from 10 years to 15 years will be 1.10×10^{-7}/ry. This delta LERF is slightly above the Region III boundary for LERF (Figure 2-1) and falls within NRC Regulatory Guide 1.174 [6] Region II ("Small Changes" in risk). Consequently, consistent with Regulatory Guide 1.174, the total Pilgrim Station LERF from internal and external events was calculated at 7.30×10^{-6}/ry to demonstrate that LERF is acceptable. This is significantly less than the Regulatory Guide 1.174 acceptance guideline of 10^{-5}/yr. (See Appendix for more details).

Other salient results from Appendix A, found the increase in risk on the combined internal and external events total integrated plant risk for those accident sequences influenced by Type A testing, given the change from a 1-in-10 years test interval to a 1-in-15 years test interval, to be 0.052% or 0.145 person-rem/ry. In addition, the change in the combined internal and external events conditional containment failure probability from 1-in-10 years to 1-in-15 years is 0.13%. A change in CCFP of less than 1% is insignificant from a risk perspective.

Therefore, incorporating external event accident sequence results into this analysis does not change the conclusion of internal events only risk assessment (i.e., increasing the Pilgrim Station ILRT interval from 10 to 15 years is an acceptable plant change from a risk perspective). These results are expected, because the proposed ILRT interval extension impacts plant risk in a very specific and limited way.

2.6 Containment Liner Corrosion Risk Impact

Recently, the NRC issued a series of Requests for Additional Information (RAIs) in response to the one-time relief requests for the ILRT surveillance interval submitted by various licensees. One of the RAIs related to the risk assessment performed in this report is provided below.

Request for Additional Information:

Inspections of reinforced and steel containments at some facilities (e.g., North Anna, Brunswick D.C. Cook, and Oyster Creek) have indicated degradation from the uninspectable (embedded) side of the steel shell and liner of primary containments. The major uninspectable areas of the Mark I containment are the vertical portion of the drywell shell and part of the shell sandwiched between the drywell floor and the basemat. Please discuss what programs are used to monitor their conditions. Also, address how potential leakage due to age-related degradation from these uninspectable areas are factored into the risk assessment in support of the requested interval extension.

The impact of the risk assessment portion of the above RAIs is summarized in this section (refer to Appendix B for further details).

The containment liner corrosion analysis utilizes the referenced Calvert Cliffs Nuclear Power Plant assessment [20] to estimate the likelihood and risk-implication of degradation-induced leakage occurring and going undetected in visual examinations during the extended test interval. It should be noted that the Calvert Cliffs analysis was performed for a concrete cylinder and dome containment with a steel liner whereas Pilgrim has a free standing steel containment building. Both sites do, however, have a concrete basemat with a steel liner.

Consistent with the Calvert Cliffs analysis, the following issues are addressed:

- Differences between the containment basemat and the drywell and torus liner
- The historical drywell/torus steel shell 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

Consistent with Calvert Cliffs analysis [20], the following six steps are performed:

1) Determine the historical liner flaw likelihood.

2) Determine aged adjusted liner flaw likelihood.

3) Determine the increase in flaw likelihood between 3, 10 and 15 years.

4) Determine the likelihood of containment breach given liner flaw.

5) Determine the visual inspection detection failure.

6) Determine the likelihood of non-detected containment leakage.

In additions to these steps, the following three additional steps are added to evaluate risk-implication of containment liner corrosion:

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

8) Evaluate the risk impact in terms of LERF.

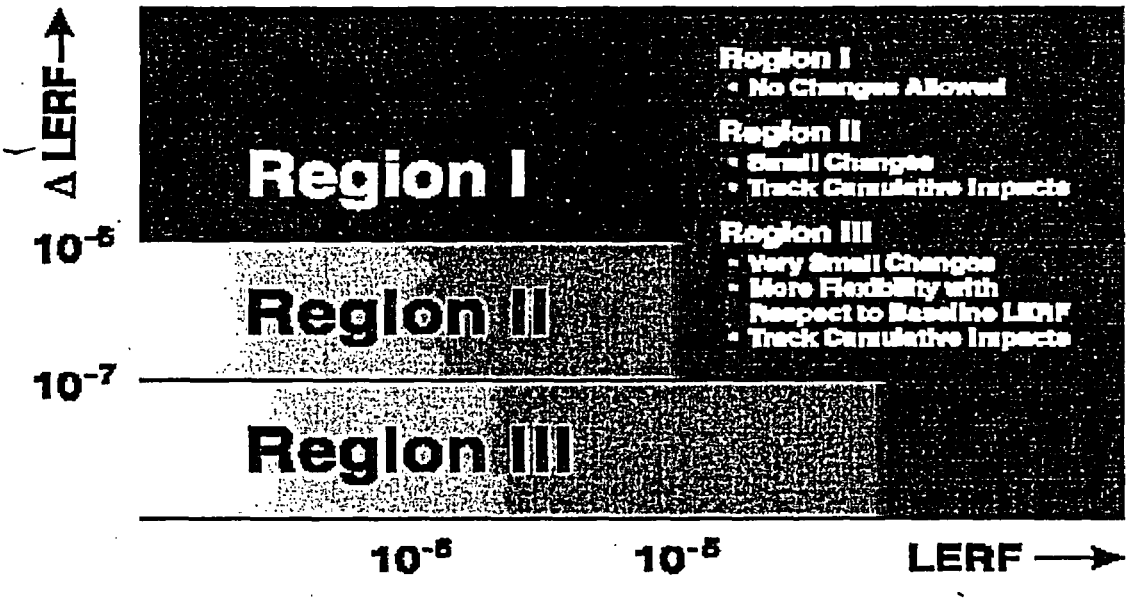
9) Evaluate the change in conditional containment failure probability.

The quantitative consideration of the containment liner corrosion analysis is discussed in more detail in Appendix B of this report. As can be seen from Appendix B, including corrosion effects in the ILRT assessment would not alter the conclusions from the original internal events analysis. That is, the change in LERF from extending the interval to 15 years from the current 10-year requirement is estimated to be 2.47×10^{-9}/ry. This value is below the NRC Regulatory Guide 1.174 [6] of 10^{-7}/yr. Therefore, because Regulatory Guide 1.174 [6] defines very small changes in LERF as below 10^{-7}/yr, increasing the ILRT interval at Pilgrim from the currently allowed 1-in-10 years to 1-in-15 years and taking into consideration the likelihood of a containment liner flaw due to corrosion is non-risk significant from a risk perspective. Additionally, the dose increase is estimated to be 2.70×10^{-3} person-rem/ry or 0.012%, and the conditional containment failure probability increase is estimated to be 0.3%. Both of these increases are also considered to be small. As a result, the ILRT interval extension is considered to have a minimal impact on plant risk (including age-adjusted corrosion impacts), and is therefore acceptable.

In addition, a series of parametric sensitivity studies (discussed in more detail in Appendix B of this report) regarding the potential age related corrosion effects on the containment steel liner also predict that even with conservative assumptions, the conclusions from the original internal events analysis would not change.

Figure 2-1

Acceptance Guidelines[8] for Large Early Release Frequency [5]



[8] The analysis will be subject to increased technical review and management attention as indicated by the darkness of the shading of the figure. In the context of the integrated decisionmaking, the boundaries between regions should not be interpreted as being definitive; the numerical values associated with defining the regions in the figure are to be interpreted as indicative values only.

Table 2-1

Pilgrim Station Internal Events Core Damage Frequency Contributions by Accident Class [7]

Class	Class Description	Point Estimate Frequency (/ry)	% Of Total CDF
I	Transients initiated sequences where the RCS is not breached and the containment integrity is not challenged prior to core melt. RCS inventory boil-off is through the SRVs to the suppression pool.	3.77×10^{-7}	5.87
II	Transients initiated sequences where containment decay heat removal systems are not available and coolant recirculation to the torus overpressurizes the containment to failure or venting. The torus is saturated.	5.86×10^{-6}	91.45
III	LOCA initiated sequences in which RCS pressure and leakage rates associated with large break LOCA's with the occurrence of early core melt. Containment integrity is maintained prior to core damage.	1.34×10^{-7}	2.09
IV	ATWS sequences at high RPV pressure and rapid containment pressurization. RCS leakage rates associated with boiloff of coolant through the cycling of SRVs/SV with early core melt subsequent to containment overpressure failure.[9]	3.39×10^{-8}	0.53
V	LOCA outside containment and failure of coolant injection, resulting in early core melting.	4.00×10^{-9}	0.06
Total Frequency		6.41×10^{-6}	1.00

[9] Due to high reactor power associated with ATWS scenarios, for these sequences containment venting capacity is insufficient to preclude overpressure failure.

Table 2-2

Summary of Pilgrim Station PSA Level 2 Containment Failures [7]

End State	Point Estimate Frequency (/ry)	% Of Total CDF
No Containment Failure	1.11×10^{-7}	1.74
Early Containment Failure[10]	1.77×10^{-7}	2.77
Late Containment Failure	6.06×10^{-6}	94.93
Bypass[11]	3.57×10^{-8}	0.56
Total Frequency	6.40×10^{-6}	100

Table 2-3

Summary of Pilgrim Station Accident Types and Their Contribution to Internal Large Early Release Frequencies [7]

Accident Type	Point Estimate Large Early Release Frequency (/ry)	% Contribution to Point Estimate Large Early Release Frequency
Station Blackout	6.43×10^{-8}	57.03
Anticipated Transient without Scram	4.49×10^{-8}	39.82
Transients	2.26×10^{-9}	2.01
Interfacing System LOCAs	1.27×10^{-9}	1.13
LOCAs	1.47×10^{-11}	0.01
Vessel Rupture	7.91×10^{-12}	0.01

[10] Excludes ATWS and ISLOCA contributions

[11] Includes ATWS and ISLOCA contributions resulting in containment bypass

Table 2-4

Summary of Pilgrim Station PSA Level 2 Containment Release Results [7]

Release Mode	Release Mode Description	Point Estimate Frequency (/ry)
CAPB-1	[CD, No VB, No CF, No CCI] Core damage occurs (CD), but the recovery of RPV injection in time prevents vessel beach (No VB). Therefore, containment integrity is not challenged (No CF) and core-concrete interactions are precluded (No CCI). However, the potential exists for some in-vessel release to the environment due to containment design leakage.	9.52×10^{-8}
CAPB-2	[CD, VB, No CF, No CCI] Core damage occurs (CD) followed by vessel breach (VB). The containment does not fail structurally and is not vented (No CF). Ex-vessel releases are recovered, therefore precluding the occurrence of core-concrete interactions (No CCI). Although the containment does not fail, vessel breach did occur, therefore the potential exists for some in- and ex-vessel releases to the environment due to containment design leakage. RPV pressure is not important because, even though high pressure induced severe accident phenomena (such as direct containment heating [DCH]) occurred, it did not fail containment.	1.27×10^{-8}
CAPB-3	[CD, VB, No CF, CCI] Core damage occurs (CD) followed by vessel breach (VB). The containment does not fail structurally and is not vented (No CF). However, ex-vessel releases are not recovered in time, and therefore core-concrete interactions occur (CCI). RPV pressure is not important because, high pressure induced severe accident phenomena even if it occurred does not significantly affect the source term as the containment does not fail nor is the vent limit reached.	2.39×10^{-9}
CAPB-4	[CD, VB, Early CF, WW, RPV pressure >200 psig at VB, No CCI] Core damage (CD) occurs followed by vessel breach (VB). The containment fails either before core damage, during core damage or at vessel breach (Early CF). The containment failure occurs in the torus (WW), above the water level. RPV pressure is greater than 200 psig at the time of vessel breach (this implies that high pressure induced severe accident phenomena [DCH] is possible). There are no core concrete interactions (No CCI) due to the present of an overlying pool of water.	3.30×10^{-9}

CD = core damage VB = vessel breach CF = containment failure
DW = drywell WW = torus RPV = reactor pressure vessel
CCI = core-concrete interactions

Table 2-4

Summary of Pilgrim Station PSA Level 2 Containment Release Results [7] (continued)

Release Mode	Release Mode Description	Point Estimate Frequency (/ry)
CAPB-5	[CD, VB, Early CF, WW, RPV pressure <200 psig at VB, No CCI] Core damage (CD) occurs followed by vessel breach (VB). The containment fails either before core damage, during core damage or at vessel breach (Early CF). The containment failure occurs in the torus (WW), above the water level. RPV pressure is less than 200 psig at the time of vessel breach; thus, precluding high pressure induced severe accident phenomena. There are no core concrete interactions (No CCI) due to the present of an overlying pool of water.	2.73×10^{-9}
CAPB-6	[CD, VB, Early CF, WW, RPV pressure >200 psig at VB, CCI] Core damage (CD) occurs followed by vessel breach (VB). The containment fails either before core damage, during core damage or at vessel breach (Early CF). The containment failure occurs in the torus (WW), above the water level. RPV pressure is greater than 200 psig at the time of vessel breach (this implies that high pressure induced severe accident phenomena [DCH] is possible). Following containment failure, core-concrete interactions occurs (CCI).	7.96×10^{-9}
CAPB-7	[CD, VB, Early CF, WW, RPV pressure <200 psig at VB, CCI]Core damage (CD) occurs followed by vessel breach (VB). The containment fails either before core damage, during core damage or at vessel breach (Early CF). The containment failure occurs in the torus (WW), above the water level. RPV pressure is less than 200 psig at the time of vessel breach; thus, precluding high pressure induced severe accident phenomena. Following containment failure, core-concrete interactions occurs (CCI).	7.94×10^{-9}
CAPB-8	[CD, VB, Early CF, DW, RPV pressure >200 psig at VB, No CCI] Core damage (CD) occurs followed by vessel breach (VB). The containment fails either before core damage, during core damage or at vessel breach (Early CF). The containment failure occurs in the drywell or below the torus water line (DW). RPV pressure is greater than 200 psig at the time of vessel breach (this implies that high pressure induced severe accident phenomena [DCH] is possible). There are no core concrete interactions (No CCI) due to the present of an overlying pool of water.	2.06×10^{-8}

CD = core damage VB = vessel breach CF = containment failure
DW = drywell WW = torus RPV = reactor pressure vessel
CCI = core-concrete interactions

Table 2-4

Summary of Pilgrim Station PSA Level 2 Containment Release Results [7] (continued)

Release Mode	Release Mode Description	Point Estimate Frequency (/ry)
CAPB-9	[CD, VB, Early CF, DW, RPV pressure <200 psig at VB, No CCI] Core damage (CD) occurs followed by vessel breach (VB). The containment fails either before core damage, during core damage or at vessel breach (Early CF). The containment failure occurs in the drywell or below the torus water line (DW). RPV pressure is less than 200 psig at the time of vessel breach (this implies that high pressure induced severe accident phenomena is precluded). There are no core concrete interactions (No CCI) due to the present of an overlying pool of water.	9.25×10^{-9}
CAPB-10	[CD, VB, Early CF, DW, RPV pressure >200 psig at VB, CCI] Core damage (CD) occurs followed by vessel breach (VB). The containment fails either before core damage, during core damage or at vessel breach (Early CF). The containment failure occurs in the drywell or below the torus water line (DW). RPV pressure is greater than 200 psig at the time of vessel breach (this implies that high pressure induced severe accident phenomena [DCH] is possible). Following containment failure, core-concrete interactions occurs (CCI).	8.54×10^{-8}
CAPB-11	[CD, VB, Early CF, DW, RPV pressure <200 psig at VB, CCI] Core damage (CD) occurs followed by vessel breach (VB). The containment fails either before core damage, during core damage or at vessel breach (Early CF). The containment failure occurs in the drywell or below the torus water line (DW). RPV pressure is less than 200 psig at the time of vessel breach; thus, precluding high pressure induced severe accident phenomena. Following containment failure, core-concrete interactions occurs (CCI).	4.35×10^{-8}
CAPB-12	[CD, VB, Late CF, WW, No CCI] Core damage (CD) occurs followed by vessel breach (VB). The containment fails late due to a loss of containment heat removal (Late CF). The containment failure occurs in the torus (WW), above the water level. RPV pressure is not important because if a high-pressure severe accident phenomena (such as DCH) occurred, it did not fail containment upon its occurrence. There are no core concrete interactions (No CCI) due to the present of an overlying pool of water.	1.70×10^{-6}

CD = core damage VB = vessel breach CF = containment failure
DW = drywell WW = torus RPV = reactor pressure vessel
CCI = core-concrete interactions

Table 2-4

Summary of Pilgrim Station PSA Level 2 Containment Release Results [7] (continued)

Release Mode	Release Mode Description	Point Estimate Frequency (/ry)
CAPB-13	[CD, VB, Late CF, WW, CCI] Core damage (CD) occurs followed by vessel breach (VB). The containment fails late (late CF) due to core-concrete interactions (CCI) after vessel breach. The containment failure occurs in the torus (WW), above the water level. RPV pressure is not important because, although a high-pressure severe accident phenomena (such as DCH) occurred, it did not fail containment.	2.30×10^{-9}
CAPB-14	[CD, VB, Late CF, DW, No CCI] urs followed by vessel breach (VB). The containment fails late due to a loss of containment heat removal (Late CF). The containment failure occurs in either the drywell or below the torus water level (DW). RPV pressure is not important, because the occurrence of a high-pressure severe accident phenomenon did not fail containment. There are no core concrete interactions (No CCI) due to the present of an overlying pool of water.	2.26×10^{-6}
CAPB-15	[CD, VB, Late CF, DW, CCI] Core damage (CD) occurs followed by vessel breach (VB). The containment fails late (late CF) due to core-concrete interactions (CCI) after vessel breach. The containment failure occurs in either the drywell or below the torus water level (DW). RPV pressure is not important because, if a high-pressure severe accident phenomenon occurred, it did not fail containment upon its occurrence.	2.12×10^{-6}
CAPB-16	[CD, VB, BYPASS, RPV pressure >200 psig, No CCI] Small break interfacing system LOCA outside containment occurs. Core damage (CD) and subsequent vessel breach (VB) results at high RPV pressure with a bypassed containment. There are no core concrete interactions (No CCI) due to the present of an overlying pool of water.	1.18×10^{-9}

CD = core damage VB = vessel breach CF = containment failure
DW = drywell WW = torus RPV = reactor pressure vessel
CCI = core-concrete interactions

Table 2-4

Summary of Pilgrim Station PSA Level 2 Containment Release Results [7] (continued)

Release Mode	Release Mode Description	Point Estimate Frequency (/ry)
CAPB-17	[CD, VB, BYPASS, RPV pressure <200 psig, No CCI] Large break interfacing system LOCA outside containment occurs. Core damage (CD) and subsequent vessel breach (VB) results at low RPV pressure with a bypassed containment. There are no core concrete interactions (No CCI) due to the present of an overlying pool of water.	6.91×10^{-9}
CAPB-18	[CD, VB, BYPASS, RPV pressure >200 psig, CCI] Small break interfacing system LOCA outside containment occurs. Core damage (CD) and subsequent vessel breach (VB) results at high RPV pressure with a bypassed containment. Following vessel breach, core-concrete interaction occurs (CCI).	4.61×10^{-10}
CAPB-19	[CD, VB, BYPASS, RPV pressure <200 psig, CCI] Large break interfacing system LOCA outside containment occurs. Core damage (CD) and subsequent vessel breach (VB) results at low RPV pressure with a bypassed containment. Following vessel breach, core-concrete interaction occurs (CCI).	2.43×10^{-8}
Total Release Frequency (CAPB-1, CAPB-2 and CAPB-3 not included)		6.30×10^{-6}
Total Frequency		6.41×10^{-6}

CD = core damage VB = vessel breach CF = containment failure
DW = drywell WW = torus RPV = reactor pressure vessel
CCI = core-concrete interactions

Table 2-5

Summary of Pilgrim Station Baseline Release Frequencies - Given EPRI TR-104285 Accident Class

EPRI Class	Class Definition	Class Description	Frequency (/ry)
1	No Containment Failure	Accident sequences in which the containment remains intact and is initially isolated. Only affected by ILRT leak testing frequency due to the incorporation of categories 3a and 3b.	9.81×10^{-8}
2	Large Containment Isolation Failures (Failure-to-close)	Accident sequences in which the containment isolation system function fails during the accident progression due to failures-to-close of large containment isolation valves (>2-inch diameter). This accident class is not affected by ILRT leak testing frequency.	4.42×10^{-11}
3a	Small Isolation Failures (Liner breach)	Accident sequences in which the containment is failed due to a pre-existing small leak in the containment structure or liner that would be identifiable only from an ILRT.	1.18×10^{-8}
3b	Large Isolation Failures (Liner Breach)	Accident sequences in which the containment is failed due to a pre-existing large leak in the containment structure or liner that would be identifiable only from an ILRT.	1.18×10^{-9}
4	Small isolation failure - failure-to-seal (Type B test)	Accident sequences in which the containment is failed due to a pre-existing failure-to-seal of Type B components that would not be identifiable from a ILRT.	Not Analyzed
5	Small isolation failure - failure-to-seal (Type C test)	Accident sequences in which the containment is failed due to a pre-existing failure-to-seal of Type C components that would not be identifiable from a ILRT.	Not Analyzed
6	Containment Isolation Failures (dependent failures, personnel errors)	Accident sequences in which the containment isolation system function fails due to "other" pre-existing failure modes not identifiable by leak rate tests (e.g., pathways left open or misalignment of containment isolation vales following a test/maintenance evolution). Not affected by ILRT leak testing frequency.	Not Analyzed
7a	Severe Accident Phenomena Induced Early Drywell Failures	Accident sequences in which vessel breach occurs and the drywell fails either before or at the time of vessel breach.	1.59×10^{-7}
7b	Severe Accident Phenomena Induced Early Torus Failures	Accident sequences in which vessel breach occurs and torus fails either before or at the time of vessel breach. Because the drywell does not fail, the entire radionuclide release passes through the torus pool.	2.19×10^{-8}
7c	Severe Accident Phenomena Induced Late Drywell Failures	Accident sequences in which vessel breach occurs, however, the drywell does not fail until a late time period.	4.38×10^{-6}
7d	Severe Accident Phenomena Induced Late Torus Failures	Accident sequences in which vessel breach occurs, however, the torus does not fail until a late time period. Because the drywell does not fail, the entire radionuclide release passes through the torus pool.	1.70×10^{-6}
8	Containment Bypassed (ATWS)	Accident sequences in which the containment is bypassed (i.e., ATWS with high power oscillations or Interfacing Systems LOCA, ISLOCA).	3.79×10^{-8}
CDF	All Level 2 CET Endstates		6.41×10^{-6}

Table 2-6

Pilgrim Station Base Case Population Dose Values for Postulated Internal Events [7 & 12]

Release Mode	Release Mode Description	Frequency (/yr)	Population Dose (50 Miles) (Person-sv)*	Population Dose Risk (PDR) (Person-rem/yr)
CAPB-1	[CD, No VB, No CF, No CCI]	9.52×10^{-8}	4.68×10^{-1}	4.46×10^{-6}**
CAPB-2	[CD, VB, No CF, No CCI]	1.27×10^{-8}	1.00×10^{2}	1.27×10^{-4}
CAPB-3	[CD, VB, No CF, CCI]	2.39×10^{-9}	1.06×10^{2}	2.53×10^{-5}
CAPB-4	[CD, VB, Early CF, WW, RPV pressure >200 psig at VB, No CCI]	3.30×10^{-9}	1.40×10^{4}	4.62×10^{-3}
CAPB-5	[CD, VB, Early CF, WW, RPV pressure <200 psig at VB, No CCI]	2.73×10^{-9}	1.82×10^{4}	4.97×10^{-3}
CAPB-6	[CD, VB, Early CF, WW, RPV pressure >200 psig at VB, CCI]	7.96×10^{-9}	1.53×10^{4}	1.22×10^{-2}
CAPB-7	[CD, VB, Early CF, WW, RPV pressure <200 psig at VB, CCI]	7.94×10^{-9}	1.69×10^{4}	1.34×10^{-2}
CAPB-8	[CD, VB, Early CF, DW, RPV pressure >200 psig at VB, No CCI]	2.06×10^{-8}	4.33×10^{4}	8.92×10^{-2}
CAPB-9	[CD, VB, Early CF, DW, RPV pressure <200 psig at VB, No CCI]	9.25×10^{-9}	2.46×10^{4}	2.28×10^{-2}
CAPB-10	[CD, VB, Early CF, DW, RPV pressure >200 psig at VB, CCI]	8.54×10^{-8}	4.53×10^{4}	3.87×10^{-1}
CAPB-11	[CD, VB, Early CF, DW, RPV pressure <200 psig at VB, CCI]	4.35×10^{-8}	3.57×10^{4}	1.55×10^{-1}
CAPB-12	[CD, VB, Late CF, WW, No CCI]	1.70×10^{-6}	9.76×10^{1}	1.66×10^{-2}
CAPB-13	[CD, VB, Late CF, WW, CCI]	2.30×10^{-9}	7.35×10^{3}	1.69×10^{-3}
CAPB-14	[CD, VB, Late CF, DW, No CCI]	2.26×10^{-6}	1.61×10^{4}	3.64
CAPB-15	[CD, VB, Late CF, DW, CCI]	2.12×10^{-6}	4.55×10^{4}	9.65
CAPB-16	[CD, VB, BYPASS, RPV pressure >200 psig, No CCI]	1.18×10^{-9}	1.89×10^{4}	2.23×10^{-3}
CAPB-17	[CD, VB, BYPASS, RPV pressure <200 psig, No CCI]	6.91×10^{-9}	5.12×10^{4}	3.54×10^{-2}
CAPB-18	[CD, VB, BYPASS, RPV pressure >200 psig, CCI]	4.61×10^{-10}	2.44×10^{4}	1.12×10^{-3}
CAPB-19	[CD, VB, BYPASS, RPV pressure <200 psig, CCI]	2.43×10^{-8}	5.66×10^{4}	1.38×10^{-1}
Total		6.41×10^{-6}	4.34×10^{5}	14.2

CD = core damage VB = vessel breach CF = containment failure
DW = drywell WW = torus RPV = reactor pressure vessel
CCI = core-concrete interactions

* 1 sv = 100 rem
** (Person-rem/yr) = (/yr) x (person-sv) x 100 (person-rem/person-sv)

Table 2-7

Pilgrim Station Population Dose Estimates As A Function of EPRI Accident Class within 50-Mile Radius

EPRI Class	Accident Class Description	Person-Rem Within 50 miles
1	No Containment Failure	1.06×10^{4}
2	Large Containment Isolation Failures (Failure-to-close)	4.53×10^{6}
3a	Small Isolation Failures (Liner breach)	1.06×10^{5}
3b	Large Isolation Failures (Liner Breach)	3.71×10^{5}
4	Small isolation failure - failure-to-seal (Type B test)	N/A
5	Small isolation failure - failure-to-seal (Type C test)	N/A
6	Containment Isolation Failures (dependent failures, personnel errors)	N/A
7a	Severe Accident Phenomena Induced Early Drywell Failures	4.53×10^{6}
7b	Severe Accident Phenomena Induced Early Torus Failures	1.82×10^{6}
7c	Severe Accident Phenomena Induced Late Drywell Failures	4.55×10^{6}
7d	Severe Accident Phenomena Induced Late Torus Failures	7.35×10^{5}
8	Containment Bypassed (ATWS)	5.66×10^{6}

Table 2-8

Pilgrim Station Dose Rates Estimates as a Function of EPRI Accident Class For Population within 50-Miles (Base Line 3 per 10 year ILRT)

EPRI Class	Accident Class Description	Person-Rem Within 50 miles	Baseline Frequency (/ry)	Dose Rate (Person-Rem/ry)
1	No Containment Failure	1.06×10^{4}	9.81×10^{-8}	1.04×10^{-3}
2	Large Containment Isolation Failures (Failure-to-close)	4.53×10^{6}	4.42×10^{-11}	2.00×10^{-4}
3a	Small Isolation Failures (Liner breach)	1.06×10^{5}	1.18×10^{-8}	1.25×10^{-3}
3b	Large Isolation Failures (Liner Breach)	3.71×10^{5}	1.18×10^{-9}	4.38×10^{-4}
4	Small isolation failure - failure-to-seal (Type B test)	N/A	N/A	N/A
5	Small isolation failure - failure-to-seal (Type C test)	N/A	N/A	N/A
6	Containment Isolation Failures (dependent failures, personnel errors)	N/A	N/A	N/A
7a	Severe Accident Phenomena Induced Early Drywell Failures	4.53×10^{6}	1.59×10^{-7}	7.20×10^{-1}
7b	Severe Accident Phenomena Induced Early Torus Failures	1.82×10^{6}	2.19×10^{-8}	3.99×10^{-2}
7c	Severe Accident Phenomena Induced Late Drywell Failures	4.55×10^{6}	4.38×10^{-6}	1.99×10^{1}
7d	Severe Accident Phenomena Induced Late Drywell Failures	7.35×10^{5}	1.70×10^{-6}	1.25×10^{0}
8	Containment Bypassed (ATWS)	5.66×10^{6}	3.79×10^{-8}	2.15×10^{-1}
	Total	2.23×10^{7}	6.41×10^{-6}	2.21×10^{1}

Table 2-9

EPRI Accident Class Frequency as a Function of ILRT Interval

EPRI Class	Baseline (3-per-10 year ILRT) /ry	Current (1-in-10 years ILRT) /ry	Proposed (1-per-15 year ILRT) /ry
1	9.81×10^{-8}	6.78×10^{-8}/ry	4.61×10^{-8}/ry
3a	1.18×10^{-8}	3.93×10^{-8}/ry	5.90×10^{-8}/ry
3b	1.18×10^{-9}	3.93×10^{-9}/ry	5.90×10^{-9}/ry

Table 2-10
Baseline Dose Rate Estimates By EPRI Accident Class for Population Within 50-Mile

EPRI Class	Accident Class Description	Dose Rate as a Function of ILRT Interval (Person-Rem/Rx Year)		
		Baseline (3-per-10 year ILRT)	Current (1-per-10 year ILRT)	Proposed (1-in-15 years ILRT)
1	No Containment Failure	1.04×10^{-3}	7.20×10^{-4}	4.89×10^{-4}
2	Large Containment Isolation Failures (Failure-to-close)	2.00×10^{-4}	2.00×10^{-4}	2.00×10^{-4}
3a	Small Isolation Failures (Liner breach)	1.25×10^{-3}	4.17×10^{-3}	6.25×10^{-3}
3b	Large Isolation Failures (Liner Breach)	4.38×10^{-4}	1.46×10^{-3}	2.19×10^{-3}
4	Small isolation failure - failure-to-seal (Type B test)	N/A	N/A	N/A
5	Small isolation failure - failure-to-seal (Type C test)	N/A	N/A	N/A
6	Containment Isolation Failures (dependent failures, personnel errors)	N/A	N/A	N/A
7a	Severe Accident Phenomena Induced Early Drywell Failures	7.20×10^{-1}	7.20×10^{-1}	7.20×10^{-1}
7b	Severe Accident Phenomena Induced Early Torus Failures	3.99×10^{-2}	3.99×10^{-2}	3.99×10^{-2}
7c	Severe Accident Phenomena Induced Late Drywell Failures	1.99×10^{1}	1.99×10^{1}	1.99×10^{1}
7d	Severe Accident Phenomena Induced Late Drywell Failures	1.25×10^{0}	1.25×10^{0}	1.25×10^{0}
8	Containment Bypassed (ATWS)	2.15×10^{-1}	2.15×10^{-1}	2.15×10^{-1}
	Total	22.128	22.132	22.134

SECTION 3

SUMMARY OF RESULTS

3.1 Internal Events Impact

An evaluation was performed to assess the risk impact of extending the current containment Type A Integrated Leak Rate Test (ILRT) interval. In performing the risk assessment evaluation, the guidance and additional information distributed by NEI in November 2001 to their Administrative Points of Contact [4, 5] regarding risk assessment evaluation of one-time extensions of containment ILRT intervals and the approach outlined in the Indian Point Unit Three Nuclear Power Plant ILRT [8, 10] extension submittal were used. The assessment also followed previous work as outline in NEI 94-01 [1], the methodology used in EPRI TR-104285 [3], and the NRC Regulatory Guide 1.174 [6].

These results demonstrate a very small impact on risk associated with the one time extension of the ILRT test interval to 15 years. The following is a brief summary of some of the key aspects of the ILRT test interval extension risk analysis:

1) The baseline (3-in-10 years) risk contribution (person-rem) associated with containment leakage affected by the ILRT and represented by Classes 3a and 3b accident scenarios is 0.0076% of the total risk.

2) When the ILRT interval is 1-in-10 years, the risk contribution of leakage (person-rem) represented by Classes 3a and 3b accident scenarios increases to 0.025% of the total risk.

3) When the ILRT interval is 1-in-15 years, the risk contribution of leakage represented by Classes 3a and 3b accident scenarios increases to 0.038% of the total risk.

4) The increase in risk on the total integrated plant risk as measured by person-rem/reactor year increases for those accident sequences influenced by Type A testing, given the change from a 1-in-10 years test interval to a 1-in-15 years test interval, is found to be 0.009% (0.002 person-rem/ry). This value can be considered to be a negligible increase in risk.

5) The risk increase in LERF from reducing the ILRT test frequency from the current once-per-l 0 years to once-per-15 years is 1.97×10^{-9}/ry. This is determined to be very small using the acceptance guidelines of Regulatory Guide 1.174.

6) The risk increase in LERF from the original 3-in-10 years test frequency; to once-per-15 years is 4.72×10^{-9}/ry. This is also found to be "very small" using the acceptance guidelines in Regulatory Guide 1.174.

7) The change in CCFP of 0.03% is deemed to be insignificant and reflects sufficient defense-in-depth.

8) Other salient results are summarized in Table 3-1. The key results to this risk assessment are those for the 10-year interval (current Pilgrim LRT interval) and the 15-year interval (proposed change). The 3-in-10 year ILRT is a baseline starting point for this risk assessment given that the pre-existing containment leakage probabilities (estimated based on industry experience - - refer to Section 1.2) are reflective of the 3-per-10 year ILRT testing.

3.2 External Events Impact

This analysis provides an evaluation of external events hazards (seismic, fires, high winds, external flooding, etc) impacts within the framework of the ILRT interval extension risk assessment. Similar to the internal events analysis, the combined impact of internal and external events confirms that the impact (due to the proposed ILRT extension) on the external hazard portion of the Pilgrim plant risk profile is comparable to that shown for internal events. It is deemed that the calculated risk increase for both internal and external hazards would remain "small".

These results demonstrate a small impact on risk associated with the one time extension of the ILRT test interval to 15 years. The following is a brief summary of some of the key aspects of the ILRT test interval extension risk analysis for the combined internal and external events analysis:

1) The baseline (3-in-10 years) risk contribution (person-rem) associated with containment leakage affected by the ILRT and represented by Classes 3a and 3b accident scenarios is 0.0336% of the total risk.

2) When the ILRT interval is 1-in-10 years, the risk contribution of leakage (person-rem) represented by Classes 3a and 3b accident scenarios increases to 0.112% of the total risk.

3) When the ILRT interval is 1-in-15 years, the risk contribution of leakage represented by Classes 3a and 3b accident scenarios increases to 0.168% of the total risk.

4) The combined internal and external events increase in risk on the total integrated plant risk for those accident sequences influenced by Type A testing, given the change from a 1-in-10 years test interval to a 1-in-15 years test interval, is found to be 0.052% (0.145person-rem/ry). This value can be considered to be a negligible increase in risk.

5) The combined internal and external events risk increase in LERF from reducing the ILRT test frequency from the current once-per-10 years to once-per-15 years is 1.10×10^{-7}/ry. This is determined to be slightly above the 10^{-7}/yr criterion of Region III, Very Small Change in Risk (Figure 2-1), of the acceptance guidelines of Regulatory Guide 1.174. Consequently, consistent with Regulatory Guide 1.174, the total Pilgrim Station LERF from internal and external events was calculated at 7.30×10^{-6}/ry to demonstrate that LERF is acceptable. This is significantly less than the Regulatory Guide 1.174 acceptance guideline of 10^{-5}/yr.

6) The combined internal and external events change in CCFP of 0.13% is deemed to be insignificant and reflects sufficient defense-in-depth.

7) Other salient results are summarized in Table 3-2.

3.3 Containment Liner Corrosion Risk Impact

This analysis provides a sensitivity evaluation of considering potential corrosion impacts within the framework of the ILRT interval extension risk assessment. The analysis confirms that the ILRT interval extension has a minimal impact on plant risk. Additionally, a series of parametric sensitivity studies regarding the potential age related corrosion effects on the steel shell also indicate that even with very conservative assumptions, the conclusions from the original analysis would not change. That is, the ILRT interval extension is judged to have a minimal impact on plant risk and is therefore acceptable.

1) The baseline (3-in-10 years) risk contribution (person-rem) associated with containment leakage affected by the ILRT and represented by Classes 3a and 3b accident scenarios is 0.0077% of the total risk.

2) When the ILRT interval is 1-in-10 years, the risk contribution of leakage (person-rem) represented by Classes 3a and 3b accident scenarios increases to 0.0259% of the total risk.

3) When the ILRT interval is 1-in-15 years, the risk contribution of leakage represented by Classes 3a and 3b accident scenarios increases to 0.0394% of the total risk.

4) The age-adjusted corrosion impact on the total integrated plant risk for those accident sequences influenced by Type A testing, given the change from a 1-in-10 years test interval to a 1-in-15 years test interval, is found to be 0.012% (0.0027person-rem/ry). This value can be considered to be a negligible increase in risk.

5) The age-adjusted corrosion impact risk increase in LERF from reducing the ILRT test frequency from the current once-per-10 years to once-per-15 years is 2.47×10^{-9}/ry. This is determined to be below the 10^{-7}/yr criterion of Region III, Very Small Change in Risk (Figure 2-1), of the acceptance guidelines of Regulatory Guide 1.174.

6) This age-adjusted corrosion impact change in CCFP of 0.03% is deemed to be insignificant and reflects sufficient defense-in-depth.

7) Other results (taken from Appendix B) of the updated ILRT assessment including the potential impact from non-detected containment leakage scenarios assuming that 100% of the leakages result in EPRI Class 3b are show in Table 3-3.

Additional sensitivity cases were also developed to gain an understanding of the containment liner corrosion sensitivity to various key parameters. The sensitivity cases are as follows:

- Sensitivity Case 1 - Flaw rate doubles every 2 years
- Sensitivity Case 2 - Flaw rate doubles every 10 years
- Sensitivity Case 3 - 5% Visual inspection failures
- Sensitivity Case 4 - 15% Visual inspection failures
- Sensitivity Case 5 - Containment breach base point 10 times lower
- Sensitivity Case 6 - Containment breach base point 10 times higher
- Sensitivity Case 7 - Flaw rate doubles every 10 years, containment breach base point 10 times lower, 5% visual inspection failures and 10% EPRI accident Class 3b are LERF (Lower bound)
- Sensitivity Case 8 - Flaw rate doubles every 2 years, containment breach base point 10 times higher, 15% visual inspection failures and 100% EPRI accident Class 3b are LERF (upper bound)

The results of the containment liner corrosion sensitivities cases, taken from Appendix B are summarized in Table 3-4.

Table 3-1

Summary of Risk Impact on Extending Type A ILRT Test Frequency – Effect of Internal Events Risk on Pilgrim ILRT Risk Assessment

	Base Case 3 Years			Extend to 10 Years			Extend to 15 Years		
EPRI Class	CDF (Per ry)	Per-Rem	Per-Rem (Per ry)	CDF (Per ry)	Per-Rem	Per-Rem (Per ry)	CDF (Per ry)	Per-Rem	Per-Rem (Per ry)
1	9.81×10^{-8}	1.06×10^{4}	1.04×10^{-3}	6.78×10^{-8}	1.06×10^{4}	7.20×10^{-4}	4.61×10^{-8}	1.06×10^{4}	4.89×10^{-4}
2	4.42×10^{-11}	4.53×10^{6}	2.00×10^{-4}	4.42×10^{-11}	4.53×10^{6}	2.00×10^{-4}	4.42×10^{-11}	4.53×10^{6}	2.00×10^{-4}
3a	1.18×10^{-8}	1.06×10^{5}	1.25×10^{-3}	3.93×10^{-8}	1.06×10^{5}	4.17×10^{-3}	5.90×10^{-8}	1.06×10^{5}	6.25×10^{-3}
3b	1.18×10^{-9}	3.71×10^{5}	4.38×10^{-4}	3.93×10^{-9}	3.71×10^{5}	1.46×10^{-3}	5.90×10^{-9}	3.71×10^{5}	2.19×10^{-3}
4	N/A	N/A	0.00	0.00	N/A	0.00	0.00	N/A	0.00
5	N/A	N/A	0.00	0.00	N/A	0.00	0.00	N/A	0.00
6	N/A	N/A	0.00	0.00	N/A	0.00	0.00	N/A	0.00
7a	1.59×10^{-7}	4.53×10^{6}	7.20×10^{-1}	1.59×10^{-7}	4.53×10^{6}	7.20×10^{-1}	1.59×10^{-7}	4.53×10^{6}	7.20×10^{-1}
7b	2.19×10^{-8}	1.82×10^{6}	3.99×10^{-2}	2.19×10^{-8}	1.82×10^{6}	3.99×10^{-2}	2.19×10^{-8}	1.82×10^{6}	3.99×10^{-2}
7c	4.38×10^{-6}	4.55×10^{6}	1.99×10^{1}	4.38×10^{-6}	4.55×10^{6}	1.99×10^{1}	4.38×10^{-6}	4.55×10^{6}	1.99×10^{1}
7d	1.70×10^{-6}	7.35×10^{5}	1.25×10^{0}	1.70×10^{-6}	7.35×10^{5}	1.25×10^{0}	1.70×10^{-6}	7.35×10^{5}	1.25×10^{0}
8	3.79×10^{-8}	5.66×10^{6}	2.15×10^{-1}	3.79×10^{-8}	5.66×10^{6}	2.15×10^{-1}	3.79×10^{-8}	5.66×10^{6}	2.15×10^{-1}
Total	6.41×10^{-6}		22.128	6.41×10^{-6}		22.132	6.41×10^{-6}		22.134
ILRT Dose Rate from 3a and 3b			1.69×10^{-3}			5.63×10^{-3}			8.44×10^{-3}
% Of Total			0.0076%			0.025%			0.038%
Delta Dose Rate from 3a and 3b (10 to 15 yr)									2.81×10^{-3}
LERF from 3b			1.18×10^{-9}			3.93×10^{-9}			5.90×10^{-9}
Delta LERF (10 to 15 yr)									1.97×10^{-9}
CCFP %			98.29%			98.33%			98.36%
Delta CCFP % (10 to 15 yr)									0.03%

Table 3-2

Summary of Risk Impact on Extending Type A ILRT Test Frequency – Effect of Internal and External Events Risk on Pilgrim ILRT Risk Assessment

	Base Case 3 Years			Extend to 10 Years			Extend to 15 Years		
EPRI Class	CDF (Per Ry)	Per-Rem	Per-Rem (Per Ry)	CDF (Per Ry)	Per-Rem	Per-Rem (Per Ry)	CDF (Per Ry)	Per-Rem	Per-Rem (Per Ry)
1	9.22×10^{-6}	1.06×10^{4}	9.78×10^{-2}	7.53×10^{-6}	1.06×10^{4}	7.98×10^{-2}	6.32×10^{-6}	1.06×10^{4}	6.70×10^{-2}
2	1.63×10^{-7}	4.53×10^{6}	7.38×10^{-1}	1.63×10^{-7}	4.53×10^{6}	7.38×10^{-1}	1.63×10^{-7}	4.53×10^{6}	7.38×10^{-1}
3a	6.60×10^{-7}	1.06×10^{5}	7.00×10^{-2}	2.20×10^{-6}	1.06×10^{5}	2.33×10^{-1}	3.30×10^{-6}	1.06×10^{5}	3.50×10^{-1}
3b	6.60×10^{-8}	3.71×10^{5}	2.45×10^{-2}	2.20×10^{-7}	3.71×10^{5}	8.17×10^{-2}	3.30×10^{-7}	3.71×10^{5}	1.22×10^{-1}
4	0.00	N/A	0.00	0.00	N/A	0.00	0.00	N/A	0.00
5	0.00	N/A	0.00	0.00	N/A	0.00	0.00	N/A	0.00
6	0.00	N/A	0.00	0.00	N/A	0.00	0.00	N/A	0.00
7a	6.82×10^{-6}	4.53×10^{6}	3.09×10^{1}	6.82×10^{-6}	4.53×10^{6}	3.09×10^{1}	6.82×10^{-6}	4.53×10^{6}	3.09×10^{1}
7b	7.47×10^{-8}	1.82×10^{6}	1.36×10^{-1}	7.47×10^{-8}	1.82×10^{6}	1.36×10^{-1}	7.47×10^{-8}	1.82×10^{6}	1.36×10^{-1}
7c	4.74×10^{-5}	4.55×10^{6}	2.16×10^{2}	4.74×10^{-5}	4.55×10^{6}	2.16×10^{2}	4.74×10^{-5}	4.55×10^{6}	2.16×10^{2}
7d	9.23×10^{-6}	7.35×10^{5}	6.79×10^{0}	9.23×10^{-6}	7.35×10^{5}	6.79×10^{0}	9.23×10^{-6}	7.35×10^{5}	6.79×10^{0}
8	4.69×10^{-6}	5.66×10^{6}	2.66×10^{1}	4.69×10^{-6}	5.66×10^{6}	2.66×10^{1}	4.69×10^{-6}	5.66×10^{6}	2.66×10^{1}
Total	7.83×10^{-5}		280.956	7.83×10^{-5}		281.159	7.83×10^{-5}		281.304
ILRT Dose Rate from 3a and 3b			9.45×10^{-2}			3.15×10^{-1}			4.72×10^{-1}
% Of Total			0.0336%			0.1120%			0.1680%
Delta Dose Rate from 3a and 3b (10 to 15 yr)									0.157
LERF from 3b			6.60×10^{-8}			2.20×10^{-7}			3.30×10^{-7}
Delta LERF (10 to 15 yr)									1.10×10^{-7}
CCFP %			87.38%			87.59%			87.72%
Delta CCFP % (10 to 15 yr)									0.13%

Table 3-3

Summary of Risk Impact on Extending Type A ILRT Test Frequency – Impact of Containment Steel Liner Corrosion on Pilgrim ILRT Intervals

	Base Case 3 Years			Extend to 10 Years			Extend to 15 Years		
EPRI Class	CDF (Per Ry)	Per-Rem	Per-Rem (Per Ry)	CDF (Per Ry)	Per-Rem	Per-Rem (Per Ry)	CDF (Per Ry)	Per-Rem	Per-Rem (Per Ry)
1	9.80×10^{-8}	1.06×10^{4}	1.04×10^{-3}	6.76×10^{-8}	1.06×10^{4}	7.16×10^{-4}	4.55×10^{-8}	1.06×10^{4}	4.83×10^{-4}
2	4.42×10^{-11}	4.53×10^{6}	2.00×10^{-4}	4.42×10^{-11}	4.53×10^{6}	2.00×10^{-4}	4.42×10^{-11}	4.53×10^{6}	2.00×10^{-4}
3a	1.18×10^{-8}	1.06×10^{5}	1.25×10^{-3}	3.93×10^{-8}	1.06×10^{5}	4.17×10^{-3}	5.90×10^{-8}	1.06×10^{5}	6.25×10^{-3}
3b	1.24×10^{-9}	3.71×10^{5}	4.60×10^{-4}	4.30×10^{-9}	3.71×10^{5}	1.59×10^{-3}	6.77×10^{-9}	3.71×10^{5}	2.51×10^{-3}
4	0.0	N/A	0.0	0.0	N/A	0.0	0.0	N/A	0.0
5	0.0	N/A	0.0	0.0	N/A	0.0	0.0	N/A	0.0
6	0.0	N/A	0.0	0.0	N/A	0.0	0.0	N/A	0.0
7b	1.59×10^{-7}	4.53×10^{6}	7.20×10^{-1}	1.59×10^{-7}	4.53×10^{6}	7.20×10^{-1}	1.59×10^{-7}	4.53×10^{6}	7.20×10^{-1}
7b	2.19×10^{-8}	1.82×10^{6}	3.99×10^{-2}	2.19×10^{-8}	1.82×10^{6}	3.99×10^{-2}	2.19×10^{-8}	1.82×10^{6}	3.99×10^{-2}
7c	4.38×10^{-6}	4.55×10^{6}	1.99×10^{1}	4.38×10^{-6}	4.55×10^{6}	1.99×10^{1}	4.38×10^{-6}	4.55×10^{6}	1.99×10^{1}
7d	1.70×10^{-6}	7.35×10^{5}	1.25×10^{0}	1.70×10^{-6}	7.35×10^{5}	1.25×10^{0}	1.70×10^{-6}	7.35×10^{5}	1.25×10^{0}
8	3.79×10^{-8}	5.66×10^{6}	2.15×10^{-1}	3.79×10^{-8}	5.66×10^{6}	2.15×10^{-1}	3.79×10^{-8}	5.66×10^{6}	2.15×10^{-1}
Total	6.41×10^{-6}		22.1568	6.41×10^{-6}		22.1606	6.41×10^{-6}		22.1633
ILRT Dose Rate from 3a and 3b			1.70×10^{-3} ($+2.45 \times 10^{-5}$)*			5.74×10^{-3} ($+1.43 \times 10^{-4}$)*			8.73×10^{-3} ($+3.34 \times 10^{-4}$)*
% Of Total			0.0077% (+0.0001%)*			0.0259% (+0.0006%)*			0.0394% (+0.0015%)*
Delta Dose Rate from 3a and 3b (10 to 15 yr)									2.70×10^{-3} (+0.0185%)*
LERF from 3b			1.24×10^{-9} ($+6.61 \times 10^{-11}$)*			4.30×10^{-9} ($+3.85 \times 10^{-10}$)*			6.77×10^{-9} ($+8.99 \times 10^{-10}$)*
Delta LERF (10 to 15 yr)									2.47×10^{-9} ($+5.14 \times 10^{-10}$)*
CCFP %			98.29% (+0.0010%)*			98.34% (+0.006%%)*			98.37% (+0.0140%)*
Delta CCFP % (10 to 15 yr)									0.03% (+0.0080%)*

* Denotes increase from original values presented in Section 2.4, Steps 7, 8, and 9 of this report.

Table 3-4

Containment Steel Liner Corrosion Sensitivity Cases

Age (Step 2)	Drywell/ Torus Breach (Step 4)	Visual Inspection & Non-Visual Flaws (Step 5)	Likelihood Flaw is LERF (EPRI Class 3b)	LERF Increase From Corrosion (3-in-10 years)	LERF Increase From Corrosion (1-in-10 years)	LERF Increase From Corrosion (1 to 15 years)	Total LERF Increase From ILRT Extension (10 to 15 years)
Base Case Doubles every 5 yrs	Base Case 1.8993%liner 0.1899%floor	Base Case 10%	Base Case 100%	Base Case 6.61×10^{-11}	Base Case 3.85×10^{-10}	Base Case 8.99×10^{-10}	Base Case 2.47×10^{-9}
Doubles every 2 yrs	Base	Base	Base	1.89×10^{-11}	3.21×10^{-10}	1.86×10^{-9}	3.50×10^{-9}
Doubles every 10 yrs	Base	Base	Base	9.83×10^{-11}	1.35×10^{-10}	1.74×10^{-10}	2.00×10^{-9}
Base	Base	5%	Base	6.32×10^{-11}	3.68×10^{-10}	8.59×10^{-10}	2.45×10^{-9}
Base	Base	15%	Base	6.90×10^{-11}	4.02×10^{-10}	9.39×10^{-10}	2.49×10^{-9}
Base	0.5090%liner[12] 0.0509%floor[12]	Base	Base	1.77×10^{-11}	1.03×10^{-10}	2.41×10^{-10}	2.09×10^{-9}
Base	7.1249% liner[13] 0.7125%floor[13]	Base	Base	2.48×10^{-10}	1.44×10^{-9}	3.37×10^{-9}	3.89×10^{-9}
Lower Bound							
Doubles every 10 yrs	0.5090%liner[12] 0.0509%floor[12]	5%	10%	2.52×10^{-12}	1.09×10^{-11}	1.99×10^{-11}	1.97×10^{-9}
Upper Bound							
Doubles every 2 yrs	7.1249% liner[13] 0.7125%floor[13]	15%	100%	7.42×10^{-11}	1.26×10^{-9}	7.31×10^{-9}	8.00×10^{-9}

[12] Base point 10 times lower than base case of 0.0001 at 20 psia.
[13] Base point 10 times higher than base case of 0.01 at 20 psia.

SECTION 4

CONCLUSIONS

4.1 Internal Events Impact

A risk assessment of the impact of changing Pilgrim Nuclear Power Station Integrated Leak Rate Test (ILRT) interval from the currently approved 1-in-10 year interval to a one-time extension to 1-in-15 years has been performed.

Based on the above results, the following are main conclusions regarding the assessment of the plant risk associated with extending the Type A ILRT test frequency from ten-years to fifteen years:

1. Regulatory Guide 1.174 [6] provides guidance for determining the risk impact of plant-specific changes to the licensing basis. Regulatory Guide 1.174 [6] 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 1-in-10 years to 1-in-15 years is 1.07×10^{-9}/ry. Since Regulatory Guide 1.174 [6] defines very small changes in LERF as below 10^{-7}/yr, increasing the ILRT interval at Pilgrim from the currently allowed one-in-ten years to one-in-fifteen years is non-risk significant from a risk perspective.

2. The increase in risk on the total integrated plant risk as measured by person-rem/reactor year increases for those accident sequences influenced by Type A testing, given the change from a 1-in-10 years test interval to a 1-in-15 years test interval, is found to be 0.009% (0.002 person-rem/ry). This value can be considered to be a negligible increase in risk.

3. The change in conditional containment failure probability (CCFP) is calculated to demonstrate the impact on 'defense-in-depth'. The $)CCFP_{10\text{-}15}$ is found to be 0.03%. This signifies a very small increase and represents a negligible change in the Pilgrim containment defense-in-depth.

Table 4-1 summarizes the above conclusions.

4.2 External Events Impact

Based on the results from Appendix A, "External Event Assessment During an Extension of the ILRT Interval," the following are main conclusions regarding the assessment of the plant risk associated with extending the Type A ILRT test frequency from ten-years to fifteen years:

1. Based on conservative methodologies in estimating the core damage frequency for seismic events and fire events, the $)LERF_{COMBINED10\text{-}15}$ of 1.07×10^{-7}/ry from extending the Pilgrim ILRT frequency from 1-in-10 years to 1-in-15 years is slightly above the 10^{-7}/yr criterion of Region III, Very Small Change in Risk (Figure 1), of the acceptance guidelines in NRC Regulatory Guide 1.174 [6]. Consequently, consistent with Regulatory Guide 1.174, the total Pilgrim Station LERF from internal and external events was calculated at 7.30×10^{-6}/ry to demonstrate that LERF is acceptable. This is less than the Regulatory Guide 1.174 acceptance guideline of 10^{-5}/yr (refer to Appendix A). Therefore, increasing the ILRT interval at Pilgrim from the currently allowed 1-in-10 years to 1-in-15 years is non-risk significant from a risk perspective.

2. The combined internal and external events increase in risk on the total integrated plant risk as measured by person-rem/reactor year increases for those accident sequences influenced by Type A testing, given the change from a 1-in-10 years test interval to a 1-in-15 years test interval, is found to be 0.050% (0.140 person-rem/yr). This value can be considered to be a negligible increase in risk.

3. The change in the combined internal and external events conditional containment failure probability from 1-in-10 years to 1-in-15 years is 0.13%. A change in)CCFP of less than 1% is insignificant from a risk perspective.

Table 4-2 summarizes the above conclusions.

4.3 Containment Liner Corrosion Risk Impact

Based on the results from Appendix B, "Risk Impact of Containment Liner Corrosion During an Extension of the ILRT Interval," the following are main conclusions regarding the assessment of the plant risk associated with extending the Type A ILRT test frequency from ten-years to fifteen years:

1. The impact of including age-adjusted corrosion effects in the ILRT assessment has minimal impact on plant risk and is therefore acceptable.

2. The change in LERF, taking into consideration the likelihood of a containment liner flaw due to age-adjusted corrosion is non-risk significant from a risk perspective. Specifically, extending the interval to 15 years from the current 10 years requirement is estimated to be about 2.47×10^{-9}/ry. This is below the Regulatory Guide 1.174 [6] acceptance criteria threshold of 10^{-7}/yr.

3. The age-adjusted corrosion impact in dose increase is estimated to be 2.70×10^{-3} person-rem/ry or 0.012% from the baseline ILRT 10 year's interval.

4. The age-adjusted corrosion impact on the conditional containment failure probability increase is estimated to be 0.3%.

5. A series of parametric sensitivity studies regarding potential age related corrosion effects on the containment steel liner also demonstrated minimal impact on plant risk.

Table 4-3 summarizes the above conclusions.

Table 4-1
Quantitative Results as a Function of ILRT Interval - Internal Events

EPRI Class	Category Description	Dose (Person-Rem Within 50 miles)	Quantitative Results as a Function of ILRT Interval			
			Current (1-per-10 year ILRT)		Proposed (1-per-15 year ILRT)	
			Accident Frequency (per ry)	Population Dose Rate (Person-Rem / Ry Within 50 miles)	Accident Frequency (per ry)	Population Dose Rate (Person-Rem / Ry Within 50 miles)
1	No Containment Failure [1]	1.06×10^{4}	6.78×10^{-8}	7.20×10^{-4}	4.61×10^{-8}	4.89×10^{-4}
2	Containment Isolation System Failure	4.53×10^{6}	4.42×10^{-11}	2.00×10^{-4}	4.42×10^{-11}	2.00×10^{-4}
3a	Small Pre-Existing Failures [1], [2]	1.06×10^{5}	3.93×10^{-8}	4.17×10^{-3}	5.90×10^{-8}	6.25×10^{-3}
3b	Large Pre-Existing Failures [1], [2]	3.71×10^{5}	3.93×10^{-9}	1.46×10^{-3}	5.90×10^{-9}	2.19×10^{-3}
4	Type B Failures (LLRT)	N/A	0.00	0.00	0.00	0.00
5	Type C Failures (LLRT)	N/A	0.00	0.00	0.00	0.00
6	Other Containment Isolation System Failure	N/A	0.00	0.00	0.00	0.00
7a	Containment Failure Due to Severe Accident (a) [3]	4.53×10^{6}	1.59×10^{-7}	7.20×10^{-1}	1.59×10^{-7}	7.20×10^{-1}
7b	Containment Failure Due to Severe Accident (b) [3]	1.82×10^{6}	2.19×10^{-8}	3.99×10^{-2}	2.19×10^{-8}	3.99×10^{-2}
7c	Containment Failure Due to Severe Accident (c) [3]	4.55×10^{6}	4.38×10^{-6}	1.99×10^{1}	4.38×10^{-6}	1.99×10^{1}
7d	Containment Failure Due to Severe Accident (d) [3]	7.35×10^{5}	1.70×10^{-6}	1.25×10^{0}	1.70×10^{-6}	1.25×10^{0}
8	Containment Bypass Accidents	5.66×10^{6}	3.79×10^{-8}	2.15×10^{-1}	3.79×10^{-8}	2.15×10^{-1}
TOTALS:			6.41×10^{-6}	22.132	6.41×10^{-6}	22.134
Increase in Dose Rate						0.009%
Increase in LERF					1.97×10^{-9}	
Increase in CCFP (%)					0.03%	

Table 4-2

Quantitative Results as a Function of ILRT Interval - Internal and External Events

EPRI Class	Category Description	Dose (Person-Rem Within 50 miles)	Quantitative Results as a Function of ILRT Interval			
			Current (1-per-10 year ILRT)		Proposed (1-per-15 year ILRT)	
			Accident Frequency (per ry)	Population Dose Rate (Person-Rem / Ry Within 50 miles)	Accident Frequency (per ry)	Population Dose Rate (Person-Rem / Ry Within 50 miles)
1	No Containment Failure [1]	1.06×10^{4}	7.53×10^{-6}	7.98×10^{-2}	6.32×10^{-6}	6.70×10^{-2}
2	Containment Isolation System Failure	4.53×10^{6}	1.63×10^{-7}	7.38×10^{-1}	1.63×10^{-7}	7.38×10^{-1}
3a	Small Pre-Existing Failures [1], [2]	1.06×10^{5}	2.20×10^{-6}	2.33×10^{-1}	3.30×10^{-6}	3.50×10^{-1}
3b	Large Pre-Existing Failures [1], [2]	3.71×10^{5}	2.20×10^{-7}	8.17×10^{-2}	3.30×10^{-7}	1.22×10^{-1}
4	Type B Failures (LLRT)	N/A	0.00	0.00	0.00	0.00
	Type C Failures (LLRT)	N/A	0.00	0.00	0.00	0.00
6	Other Containment Isolation System Failure	N/A	0.00	0.00	0.00	0.00
7a	Containment Failure Due to Severe Accident (a) [3]	4.53×10^{6}	6.82×10^{-6}	3.09×10^{1}	6.82×10^{-6}	3.09×10^{1}
7b	Containment Failure Due to Severe Accident (b) [3]	1.82×10^{6}	7.47×10^{-8}	1.36×10^{-1}	7.47×10^{-8}	1.36×10^{-1}
7c	Containment Failure Due to Severe Accident (c) [3]	4.55×10^{6}	4.74×10^{-5}	2.16×10^{2}	4.74×10^{-5}	2.16×10^{2}
7d	Containment Failure Due to Severe Accident (d) [3]	7.35×10^{5}	9.23×10^{-6}	6.79×10^{0}	9.23×10^{-6}	6.79×10^{0}
8	Containment Bypass Accidents	5.66×10^{6}	4.69×10^{-6}	2.66×10^{1}	4.69×10^{-6}	2.66×10^{1}
TOTALS:			7.83×10^{-5}	279.586	7.83×10^{-5}	279.727
Increase in Dose Rate						0.052%
Increase in LERF					1.10×10^{-7}	
Increase in CCFP (%)					0.13%	

Table 4-3
Quantitative Results as a Function of ILRT Interval - Liner Corrosion Impact

EPRI Class	Category Description	Dose (Person-Rem Within 50 miles)	Quantitative Results as a Function of ILRT Interval			
			Current (1-per-10 year ILRT)		Proposed (1-per-15 year ILRT)	
			Accident Frequency (per ry)	Population Dose Rate (Person-Rem / Ry Within 50 miles)	Accident Frequency (per ry)	Population Dose Rate (Person-Rem / Ry Within 50 miles)
1	No Containment Failure [1]	1.06×10^{4}	6.76×10^{-8}	7.16×10^{-4}	4.55×10^{-8}	4.83×10^{-4}
2	Containment Isolation System Failure	4.53×10^{6}	4.42×10^{-11}	2.00×10^{-4}	4.42×10^{-11}	2.00×10^{-4}
3a	Small Pre-Existing Failures [1], [2]	1.06×10^{5}	3.91×10^{-8}	4.15×10^{-3}	5.87×10^{-8}	6.22×10^{-3}
3b	Large Pre-Existing Failures [1], [2]	3.71×10^{5}	4.30×10^{-9}	1.59×10^{-3}	6.77×10^{-9}	2.51×10^{-3}
4	Type B Failures (LLRT)	N/A	0.0	0.0	0.0	0.0
5	Type C Failures (LLRT)	N/A	0.0	0.0	0.0	0.0
6	Other Containment Isolation System Failure	N/A	0.0	0.0	0.0	0.0
7a	Containment Failure Due to Severe Accident (a) [3]	4.53×10^{6}	1.59×10^{-7}	7.19×10^{-1}	1.59×10^{-7}	7.19×10^{-1}
7b	Containment Failure Due to Severe Accident (b) [3]	1.82×10^{6}	2.19×10^{-8}	3.99×10^{-2}	2.19×10^{-8}	3.99×10^{-2}
7c	Containment Failure Due to Severe Accident (c) [3]	4.55×10^{6}	4.38×10^{-6}	1.99×10^{1}	4.38×10^{-6}	1.99×10^{1}
7d	Containment Failure Due to Severe Accident (d) [3]	7.35×10^{5}	1.70×10^{-6}	1.25×10^{0}	1.70×10^{-6}	1.25×10^{0}
8	Containment Bypass Accidents	5.66×10^{6}	3.79×10^{-8}	2.15×10^{-1}	3.79×10^{-8}	2.15×10^{-1}
TOTALS:			6.41×10^{-6}	22.1606	$6.41f \times 10^{-6}$	22.1633
Increase in Dose Rate						0.012%
Increase in LERF					2.47×10^{-9}	
Increase in CCFP (%)					0.30%	

Notes to Tables 15, 16, and 17:

1) Only EPRI categories 1, 3a, and 3b are affected by ILRT (Type A) interval changes.

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

3) EPRI Class 7, containment failure due to severe accident, was subdivided into four subgroups based on Pilgrim Level 2 containment failure modes for dose allocation purposes. Note that this EPRI class is not affected by ILRT interval changes.

SECTION 5

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(16) United States Nuclear Regulatory Commission, "Procedural and Submittal Guidance for the Individual Plant Examination of External Events (IPEEE) for Severe Accident Vulnerabilities," NUREG-1407, June 1991.

(17) USNRC, "PRA Procedure Guide - A Guide to the Performance of Probabilistic Risk Assessments for Nuclear Power Plants", NUREG/CR-2300, January 1983. .

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(20) Response to Request for Additional Information Concerning the License Amendment Request for a One-Time Integrated Leakage Rate Test Extension, Letter from Mr. C. H. Cruse (Calvert Cliffs Nuclear Power Plant) to NRC Document Control Desk, March 27, 2002.

(21) General Physics Corporation Report GP-R-66195010, "Reactor Containment Building Integrated Leakage Rate Test Final Report".

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Appendix A

External Event Assessment During an Extension of the ILRT Interval

Table of Contents

Tables

A1.0 Introduction

This appendix discusses the risk-implication associated with external hazards in support of the Pilgrim Station Integrated Leak Rate Testing (ILRT) interval extension risk assessment.

In response to Generic Letter 88-20, Supplement 4 [14], Pilgrim submitted an Individual Plant Examination of External Events (IPEEE) in July 1994 [15]. The IPEEE was a review of external hazard risk (i.e., seismic, fires, high winds, external flooding, etc) to identify potential plant vulnerabilities and to understand severe accident risks. The results of the Pilgrim Station IPEEE are therefore used in this risk assessment to provide a comparison of the effect of external hazards when extending the current 1-in-10 years to 1-in-15 years Type A ILRT interval.

A2.0 Pilgrim IPEEE Seismic Analysis

A2.1 Seismic Analysis Methodology Selection

The Pilgrim plant has been designed to accommodate a safe-shutdown earthquake (SSE) with 0.15g-peak ground acceleration. The seismic analysis performed in the IPEEE study is intended to act as a performance check on the design, estimating seismic capacity beyond the SSE.

The seismic analysis methodology implemented for Pilgrim satisfied the NRC requirements for performing a seismic IPEEE as presented in Generic Letter 88-20, Supplement 4 [14]. The methodology comprises a Seismic Probabilistic Risk Assessment (SPRA) developed in accordance with the guidance provided in NUREG-1407 [16] and NUREG/CR-2300 [17]. The SPRA logic model was developed using a fault tree linking approach similar to the Internal Events IPE. This approach permits the explicit modeling of system/component dependencies that exist between event tree top events. The SPRA also includes a simplified containment performance model, which was developed to address scenarios leading to significant early containment releases during a seismic event.

A2.2 Seismic Analysis Conclusions

The conclusions of the Pilgrim IPEEE seismic risk analysis [15] are as follows:

1. The Pilgrim seismic CDF is 5.82×10^{-5} /yr.

2. The median capacity of the Pilgrim Station plant is 0.48g PGA, which is approximately 3.2 times the Safe Shutdown Earthquake level of 0.15g.

3. The overall plant HCLPF (High Confidence Low Probability of Failure) capacity at Pilgrim is 0.25g PGA. (The plant HCLPF provides a measure of the seismic structural integrity of structures and equipment.)

4. Ground motions greater than 0.25g PGA dominate the Seismic CDF. PGA levels greater than the plant median capacity of 0.48g contribute approximately 42 percent of the CDF.

5. The total mean frequency of early release is x 1.59×10^{-5}/yr.

A3.0 Pilgrim IPEEE Fire Analysis

A3.1 Fire Analysis Methodology Selection

The Fire analysis performed for the Pilgrim Station IPEEE submittal [15] use the EPRI Fire PRA methodology [19] following the guidance of NUREG-1407 [16]. The fire PRA analysis entailed the identification of critical areas of vulnerability, the calculation of fire initiation frequencies, the identification of fire-induced initiating events and their impact on systems, the disabling of critical safety functions, and potential fire-induced containment failure. The core damage frequency (CDF) contribution due to internal fires was calculated as 2.2×10^{-5}/ry [15].

A3.2 Fire Analysis Conclusions

The conclusions of the Pilgrim Station IPEEE fire PRA [15] are as follows:

1. Important fire sequences are functionally similar to the important internal event sequences. This analysis further supports the IPE insights as to the importance of support systems such as AC power, TBCCW, RBCCW, and SSW.

2. The results show that the fire risk does not present a significant contributor to the overall plant risk. The results also show that Pilgrim Station does not contain any significant vulnerabilities or "outliers" in the fire risk.

3. Factors that fires do not present a significant risk contributor are based on the following:

   - Pilgrim Station meets Appendix R and Appendix A requirements for spatial requirements and redundant capabilities.

   - Pilgrim has an effective transient combustible control program and an effective program of inspecting and maintaining fire barriers.

4. No additional containment vulnerabilities resulting from fire and random equipment failures were seen.

A4.0 Other External Hazards

The Pilgrim Station IPEEE submittal [15], in addition to the internal fires and seismic events, examined a number of other external hazards:

- High Winds and /Tornadoes
- External Flooding
- Ice, Hazardous Chemical, Transportation and Nearby Facility Incidents

No risks to the plant occasioned by high winds and tornadoes, external floods, ice, and hazardous chemical, transportation and nearby facility incidents were identified that might lead to core damage with a predicted frequency in excess of 10^{-6}/year. Therefore, these other external event hazards are not included in this appendix and are expected not to impact the conclusions of this ILRT risk assessment.

A5.0 Effect of External Events Hazard Risk on ILRT Risk Assessment

A5.1 Assumptions

1) The baseline 50-mile population person-rem for both seismic and fire induced EPRI accident class is base on the baseline 3-per-10 year ILRT internal events EPRI class person-rem value as presented in Table 2-10 (page 62-of-77).

2) All seismic-induced release categories are considered to occur from the drywell. This is to be consistent with the Pilgrim Station IPEEE [15] reported results, which did not provide a specific containment release location.

3) Because the Pilgrim Station IPEEE [15] did not report any LERF accident progression releases, a conservative LERF contribution that approximates 10% of external events CDF is assumed. (Note: the Pilgrim Station internal events LERF versus CDF relationship are approximately 1.76%).

A5.2 Inputs

1) In order to support the Severe Accident Mitigation Alternatives (SAMA) evaluation for the Pilgrim Station license extension, the Pilgrim Station IPEEE submittal [15] for the seismic induced core damage scenarios was revised [12 & 18]. The results of the revised Pilgrim Station seismic risk core damage and plant damage states profiles are presented in Tables A-1 and A-2, respectively. This information is used in this appendix to provide insight into the impact of external hazard risk on the conclusions of this ILRT risk assessment.

2) The seismic-induced EPRI accident classes are based on the binding scheme presented in Table A-3. Other severe accidents such as intact containment leakage and containment bypass are accounted for in other EPRI categories.

3) In order to support the SAMA evaluation for the Pilgrim Station license extension, the Pilgrim Station IPEEE submittal [15] for the fire PRA induced core damage scenarios was revised [12 & 18]. The results the revised Pilgrim Station fire PRA risk core damage and plant damage states profiles are presented in Tables A-4, A-5 and A-6 respectively. This information is used in this appendix to provide insight into the impact of external hazard risk on the conclusions of this ILRT risk assessment.

4) The fire-induced EPRI accident classes are based on the binding scheme presented in Table A-7. Other severe accidents such as intact containment leakage and containment bypass are accounted for in other EPRI categories.

5) Based on the revised seismic and fire initiators, the Pilgrim Station external event initiated CDF is approximately 1.91×10^{-5}/ry (internal fires) + 5.28×10^{-5}/ry (seismic) = 7.19×10^{-5}/ry.

A5.3 Method of Analysis

The Pilgrim Station IPEEE external events risk information presented in Sections A2, A3 and A4 is used to calculate, in accordance with the NEI Interim Guidance [4, 5] the following:

1) Evaluate the risk impact for the New Surveillance Intervals of Interest

2) Evaluate the external hazard risk impact in terms of LERF

3) Evaluate the external hazard change in conditional containment failure probability

Evaluate the risk impact for the New Surveillance Intervals of Interest.

This step calculates the percentage of the total dose rate attributable to EPRI accident Classes 3a and 3b (those accident classes affected by change in ILRT surveillance interval) and the change in this result dose rate from the base dose rate attributable to changes in ILRT surveillance interval.

As discussed in Section 2.4.3, Step 3 of this report (see page 32 of 77), the frequency per year for EPRI Category 3a and 3b are calculated as:

$$EX\_CLASS\_3a\_FREQ_{3\text{-}10} = PROB_{class\_3a\_3\text{-}10} * [CDF - (CDF_{LERF} + CDF_{NO\_LERF})]$$
$$EX\_CLASS\_3b\_FREQ_{3\text{-}10} = PROB_{class\_3b\_3\text{-}10} * [CDF - (CDF_{LERF} + CDF_{NO\_LERF})]$$
$$EX\_CLASS\_3a\_FREQ_{1\text{-}10} = PROB_{class\_3a\_1\text{-}10} * [CDF - (CDF_{LERF} + CDF_{NO\_LERF})]$$
$$EX\_CLASS\_3b\_FREQ_{1\text{-}10} = PROB_{class\_3b\_1\text{-}10} * [CDF - (CDF_{LERF} + CDF_{NO\_LERF})]$$
$$EX\_CLASS\_3a\_FREQ_{1\text{-}15} = PROB_{class\_3a\_1\text{-}15} * [CDF - (CDF_{LERF} + CDF_{NO\_LERF})]$$
$$EX\_CLASS\_3b\_FREQ_{1\text{-}15} = PROB_{class\_3b\_1\text{-}15} * [CDF - (CDF_{LERF} + CDF_{NO\_LERF})]$$

Where:

$EX\_CLASS\_3a\_FREQ_{3\text{-}10}$ = external events frequency of small pre-existing containment liner leakage given a 3-in-10 years ILRT interval

$EX\_CLASS\_3b\_FREQ_{3\text{-}10}$ = external events frequency of large pre-existing containment liner leakage given a 3-in-10 years ILRT interval

$EX\_CLASS\_3a\_FREQ_{1\text{-}10}$ = external events frequency of small pre-existing containment liner leakage given a 1-in-10 years ILRT interval

$EX\_CLASS\_3b\_FREQ_{1\text{-}10}$ = external events frequency of large pre-existing containment liner leakage given a 1-in-10 years ILRT interval

$EX\_CLASS\_3a\_FREQ_{1\text{-}15}$ = external events frequency of small pre-existing containment liner leakage given a 1-in-15 years ILRT interval

$EX\_CLASS\_3b\_FREQ_{1\text{-}15}$ = external events frequency of large pre-existing containment liner leakage given a 1-in-15 years ILRT interval

$PROB_{class\_3a\_3\text{-}10}$ = probability of small pre-existing containment liner leakage
= 0.027 [Section 2.3, input #8]

$PROB_{class\_3b\_3\text{-}10}$ = probability of large pre-existing containment liner leakage
= 0.0027 [Section 2.3, input #9]

$PROB_{class\_3a\_1\text{-}10}$ = probability of small pre-existing containment liner leakage
= 0.090 [Section 2.4.5, Step 5, page 37 of 77]

$PROB_{class\_3b\_1\text{-}10}$ = probability of large pre-existing containment liner leakage
= 0.0090 [Section 2.4.5, Step 5, page 37 of 77]

$PROB_{class\_3a\_1\text{-}15}$ = probability of small pre-existing containment liner leakage
= 0.135 [Section 2.4.5, Step 5, page 37 of 77]

$PROB_{class\_3b\_1\text{-}15}$ = probability of large pre-existing containment liner leakage
= 0.0135 [Section 2.4.5, Step 5, page 38 of 77]

CDF = the Pilgrim Station external events initiated CDF is approximately 1.91×10^{-5}/ry (internal fires) + 5.28×10^{-5}/ry (seismic) = 7.19×10^{-5}/ry [Section A5.2, Input#5].

Based on the previous discussion in Section 2.4.1, Step 1, of this calculation, the following external event accident scenarios are excluded from the 3a and 3b frequency calculation because they cannot result in a LERF release or independently result in LERF:

- Fire-induced early release scenarios (4.36E-07/ry)
  FCAPB-4 + FCAPB-5 + ... FCAPB-11 — Table A-6
  3.16E-09 + 3.33E-09 + 1.82E-08 + 2.81E-08 + 2.97E-08 + 1.72E-08 + 1.89E-07 + 1.47E-07 = 4.36E-07/ry

- Fire-induced loss of decay heat removal scenarios (1.80E-05/ry)
  Fire Class IIA + Fire Class IIB + Fire Class IIC + Fire Class IID + Fire Class IIE — Table A-5
  4.81E-06 + 7.05E-06 + 8.06E-08 + 6.07E-06 + 1.76E-08 = 1.80E-05/ry

- Seismic-induced early release scenarios (1.09E-05/ry)
  L2LSISOL + L2QUSTRX + L2SCFE + L2CONTFL — Table A-2
  1.63E-07 + 2.47E-06 + 3.62E-06 + 4.66E-06 = 1.09E-05/ry

- Seismic-induced loss of decay heat removal scenarios (1.74E-05/ry)
  Seismic Class IIA + Seismic Class IIB — Table A-1
  1.74E-05 + 5.28E-09 = 1.74E-05/ry

- Wide-spread failure of seismic safe shutdown SSCs (1.10E-06/ry)
  SURR-CFE + SURR-CFL — Table A-2
  1.90E-07 + 9.09E-07 = 1.10E-06/ry

Therefore, the baseline frequency of category 3a due to external events is calculated as

$$EX\_CLASS\_3a\_FREQ_{3\text{-}10} = 0.027*[7.19E\text{-}5 - (4.36E\text{-}07 + 1.09E\text{-}05 + 1.80E\text{-}05 + 1.74E\text{-}05 + 1.10E\text{-}06)]$$

$$EX\_CLASS\_3a\_FREQ_{3\text{-}10} = 6.50E\text{-}7/ry$$

$$EX\_CLASS\_3a\_FREQ_{1\text{-}10} = 0.090*[7.19E\text{-}5 - (4.36E\text{-}07 + 1.09E\text{-}05 + 1.80E\text{-}05 + 1.74E\text{-}05 + 1.10E\text{-}06)]$$

$$EX\_CLASS\_3a\_FREQ_{1\text{-}10} = 2.16E\text{-}6/ry$$

$$EX\_CLASS\_3a\_FREQ_{1\text{-}15} = 0.135*[7.19E\text{-}5 - (4.36E\text{-}07 + 1.09E\text{-}05 + 1.80E\text{-}05 + 1.74E\text{-}05 + 1.10E\text{-}06)]$$

$$EX\_CLASS\_3a\_FREQ_{1\text{-}15} = 3.24E\text{-}6/ry$$

Similarly, the baseline frequency of category 3b due to external events is calculated as

$EX\_CLASS\_3b\_FREQ_{3\text{-}10} = 0.0027*[7.19E\text{-}5 - (4.36E\text{-}07 + 1.09E\text{-}05 + 1.80E\text{-}05 + 1.74E\text{-}05 + 1.70E\text{-}06)]$

$EX\_CLASS\_3b\_FREQ_{3\text{-}10} = 6.50E\text{-}8/ry$

$EX\_CLASS\_3b\_FREQ_{1\text{-}10} = 0.0090*[7.19E\text{-}5 - (4.36E\text{-}07 + 1.09E\text{-}05 + 1.80E\text{-}05 + 1.74E\text{-}05 + 1.10E\text{-}06)]$

$EX\_CLASS\_3b\_FREQ_{1\text{-}10} = 2.16E\text{-}7/ry$

$EX\_CLASS\_3b\_FREQ_{1\text{-}15} = 0.0135*[7.19E\text{-}5 - (4.36E\text{-}07 + 1.09E\text{-}05 + 1.80E\text{-}05 + 1.74E\text{-}05 + 1.10E\text{-}06)]$

$EX\_CLASS\_3b\_FREQ_{1\text{-}15} = 3.24E\text{-}7/ry$

Increase to EPRI class 1 frequencies

$EX\_CLASS\_1\_FREQ_{3\text{-}10} = EX\_NCF - EX\_CLASS\_3a\_FREQ_{3\text{-}10} - EX\_CLASS\_3b\_FREQ_{3\text{-}10}$
$EX\_CLASS\_1\_FREQ_{1\text{-}10} = EX\_NCF - EX\_CLASS\_3a\_FREQ_{1\text{-}10} - EX\_CLASS\_3b\_FREQ_{1\text{-}10}$
$EX\_CLASS\_1\_FREQ_{1\text{-}15} = EX\_NCF - EX\_CLASS\_3a\_FREQ_{1\text{-}15} - EX\_CLASS\_3b\_FREQ_{1\text{-}15}$

Where:

$EX\_CLASS\_1\_FREQ_{3\text{-}10}$ = external events frequency of EPRI Class 1 given a 3-in-10 years ILRT interval

$EX\_CLASS\_1\_FREQ_{1\text{-}10}$ = external events frequency of EPRI Class 1 given a 1-in-10 years ILRT interval

$EX\_CLASS\_1\_FREQ_{1\text{-}15}$ = external events frequency of EPRI Class 1 given a 1-in-15 years ILRT interval

$EX\_CLASS\_3a\_FREQ_{3\text{-}10}$ = external events frequency of small pre-existing containment liner leakage given a 3-in-10 years ILRT interval
= 6.50E-7/ry [Above write-up, page A-6 of A-29]

$EX\_CLASS\_3b\_FREQ_{3\text{-}10}$ = external events frequency of large pre-existing containment liner leakage given a 3-in-10 years ILRT interval
= 6.50E-8/ry [Above write-up, page A-7 of A-29]

$EX\_CLASS\_3a\_FREQ_{1\text{-}10}$ = external events frequency of small pre-existing containment liner leakage given a 1-in-10 years ILRT interval
= 2.16E-6/ry [Above write-up, page A-6 of A-29]

$EX\_CLASS\_3b\_FREQ_{1\text{-}10}$ = external events frequency of large pre-existing containment liner leakage given a 1-in-10 years ILRT interval
= 2.16E-7/ry [Above write-up, page A-7 of A-29]

$EX\_CLASS\_3a\_FREQ_{1\text{-}15}$ = external events frequency of small pre-existing containment liner leakage given a 1-in-15 years ILRT interval
= 3.24E-6/ry [Above write-up, page A-7 of A-29]

EX_CLASS_3b_FREQ$_{1\text{-}15}$ = external events frequency of large pre-existing containment liner leakage given a 1-in-15 years ILRT interval
= 3.24E-7/ry [Above write-up, page A-7 of A-29]

EX_NCF = FCAPB-1 + FCAPB + FCAPB + NCF$_{SEISMIC}$ + SURR-NCF

Where:

EX_NCF = external events no containment failure frequency
FCAPB-1 = frequency of fire collapsed accident progression bin 1 = 1.06E-7/ry [Table A-6]
FCAPB-2 = frequency of fire collapsed accident progression bin 2 = 1.23E-8/ry [Table A-6]
FCAPB-3 = frequency of fire collapsed accident progression bin 3 = 1.55E-9/ry [Table A-6]
NCF$_{SEISMIC}$ = seismic event no containment failure frequency = 9.19E-6 [Table A-2]
SURR-NCF = seismic safe shutdown SSCs (surrogate element) no containment failure frequency
= 5.33E-7 [Table A-2]

Therefore:

EX_NCF = 1.06E-7 + 1.23E-8 + 1.55E-9 + 9.19E-6 + 5.33E-7
EX_NCF = 9.84E-6/ry

Therefore,

EX_CLASS_1_ FREQ$_{3\text{-}10}$ = 9.84E-6/ry - 6.50E-7/ry - 6.50E-8/ry = 9.12E-6
EX_CLASS_1_ FREQ$_{1\text{-}10}$ = 9.84E-6/ry - 2.16E-6/ry - 2.16E-7/ry = 7.45E-6
EX_CLASS_1_ FREQ$_{1\text{-}15}$ = 9.84E-6/ry - 3.24E-6/ry - 3.24E-7/ry = 6.27E-6

The change in population dose rate is calculated as outline in Section 2.4.7, Step 7 of this calculation (see page 30 of 54). The results of this calculations when using the information contain in Section A5.1 and Section A5.2, is presented below as follows:

For 3-in-10 years (internal fires and seismic event),

EPRI Class	Person-rem	Frequency/Ry	Person-rem/Ry
1	1.06×10^{4}	9.12×10^{-6}	9.67×10^{-2}
2	4.53×10^{6}	1.63×10^{-7}	7.37×10^{-1}
3a	1.06×10^{5}	6.50×10^{-7}	6.88×10^{-2}
3b	3.71×10^{5}	6.50×10^{-8}	2.41×10^{-2}
4	N/A	0.00	0.00
5	N/A	0.00	0.00
6	N/A	0.00	0.00
7a	4.53×10^{6}	6.66×10^{-6}	3.02×10^{1}
7b	1.82×10^{6}	5.28×10^{-8}	9.61×10^{-2}
7c	4.55×10^{6}	4.30×10^{-5}	1.96×10^{2}
7d	7.35×10^{5}	7.53×10^{-6}	5.54×10^{-0}
8	5.66×10^{6}	4.66×10^{-6}	2.63×10^{1}
Total		**7.19×10^{-5}**	**258.800**

For 1-in-10 years (internal fires and seismic event),

EPRI Class	Person-rem	Frequency/Ry	Person-rem/Ry
1	1.06×10^{4}	7.45×10^{-6}	7.91×10^{-2}
2	4.53×10^{6}	1.63×10^{-7}	7.37×10^{-1}
3a	1.06×10^{5}	2.16×10^{-6}	2.29×10^{-1}
3b	3.71×10^{5}	2.16×10^{-7}	8.02×10^{-2}
4	N/A	0.00	0.00
5	N/A	0.00	0.00
6	N/A	0.00	0.00
7a	4.53×10^{6}	6.66×10^{-6}	3.02×10^{1}
7b	1.82×10^{6}	5.28×10^{-8}	9.61×10^{-2}
7c	4.55×10^{6}	4.30×10^{-5}	1.96×10^{2}
7d	7.35×10^{5}	7.53×10^{-6}	5.54×10^{-0}
8	5.66×10^{6}	4.66×10^{-6}	2.63×10^{1}
Total		**7.19×10^{-5}**	**258.998**

For 1-in-15 years (internal fires and seismic event),

EPRI Class	Person-rem	Frequency/Ry	Person-rem/Ry
1	1.06×10^{4}	6.27×10^{-6}	6.65×10^{-2}
2	4.53×10^{6}	1.63×10^{-7}	7.37×10^{-1}
3a	1.06×10^{5}	3.24×10^{-6}	3.44×10^{-1}
3b	3.71×10^{5}	3.24×10^{-7}	1.20×10^{-1}
4	N/A	0.00	0.00
5	N/A	0.00	0.00
6	N/A	0.00	0.00
7a	4.53×10^{6}	6.66×10^{-6}	3.02×10^{1}
7b	1.82×10^{6}	5.28×10^{-8}	9.61×10^{-2}
7c	4.55×10^{6}	4.30×10^{-5}	1.96×10^{2}
7d	7.35×10^{5}	7.53×10^{-6}	5.54×10^{-0}
8	5.66×10^{6}	4.66×10^{-6}	2.63×10^{1}
Total		**7.19×10^{-5}**	**259.141**

Based on the results summarized above and those presented in Table 2-10 (see page 62 of 77), for the current Pilgrim Station 1-in10 years ILRT interval, the percentage contribution to total dose rate from EPRI's accident Classes 3a and 3b is calculated as follows:

$$PER\_CHG_{COMBINED\text{-}10} = \left[\frac{CLASS\_3a\_DOSE_{COMBINED\text{-}10} + CLASS\_3b\_DOSE_{COMBINED\text{-}10}}{TOT\text{-}DOSE_{COMBINED\text{-}10}} \right] * 100$$

Where:

$PER\_CHG_{COMBINED\text{-}10}$ = combined internal and external events percentage contribution to total dose rate from EPRI's accident Classes 3a and 3b given an 1-in-10 years ILRT interval

$CLASS\_3a\_DOSE_{COMBINED\text{-}10}$ = combined internal and external events EPRI accident Class 3a dose

rate given a 1-in-10 years ILRT interval

= $CLASS\_3a\_DOSE_{INTERNAL-10}$ + $CLASS\_3a\_DOSE_{EXTERNAL-10}$

$CLASS\_3b\_DOSE_{COMBINED-10}$ = combined internal and external events EPRI accident Class 3b dose rate given a 1-in-10 years ILRT interval

= $CLASS\_3b\_DOSE_{INTERNAL-10}$ + $CLASS\_3b\_DOSE_{EXTERNAL-10}$

$CLASS\_3a\_DOSE_{INTERNAL-10}$ = internal events EPRI accident Class 3a dose rate given a 1-in-10 years ILRT interval = 4.17×10^{-3} /ry [Table 2-10]

$CLASS\_3b\_DOSE_{INTERNAL-10}$ = internal events EPRI accident Class 3b dose rate given a 1-in-10 years ILRT interval = 1.46×10^{-3} /ry [Table 2-10]

$CLASS\_3a\_DOSE_{EXTERNAL-10}$ = external events EPRI accident Class 3a dose rate given a 1-in-10 years ILRT interval = 2.29×10^{-1} person-rem/ry [See for 1-in-10 years table above]

$CLASS\_3b\_DOSE_{EXTERNAL\ 10}$ = external events EPRI accident Class 3b dose rate given a 1-in-10 years ILRT interval = 8.02×10^{-2} person-rem/ry [See for 1-in-10 years table above]

$TOT\text{-}DOSE_{COMBINED-10}$ = Total combined internal and external events dose rate for all EPRI's Classes given a 1-in-10 years ILRT interval

= $TOT\text{-}DOSE_{INTERNAL-10}$ + $TOT\text{-}DOSE_{EXTERNAL-10}$

$TOT\text{-}DOSE_{INTERNAL-10}$ = Total internal events dose rate for all EPRI's Classes given a 1-in-10 years ILRT interval = 22.132 (person-rem/ry) [Table 2-10]

$TOT\text{-}DOSE_{EXTERNAL-10}$ = Total external events dose rate for all EPRI's Classes given a 1-in-10 years ILRT interval = 258.998 (person-rem/ry) [See for 1-in-10 years table above]

Therefore,

$$PER\_CHG_{COMBINED-10} = \left[\frac{(4.17 \times 10^{-3} + 2.29 \times 10^{-1}) + (1.46 \times 10^{-3} + 8.02 \times 10^{-2})}{22.132 + 258.998} \right] * 100$$

$PER\_CHG_{COMBINED-10}$ = 0.1120%

The percentage contribution to total dose rate from EPRI's accident Classes 3a and 3b based on the proposed 1-in-15 years ILRT interval is calculated as follows:

$$PER\_CHG_{COMBINED-15} = \left[\frac{CLASS\_3a\_DOSE_{COMBINED-15} + CLASS\_3b\_DOSE_{COMBINED-15}}{TOT\text{-}DOSE_{COMBINED-15}} \right] * 100$$

Where:

$PER\_CHG_{COMBINED\text{-}15}$ = combined internal and external events percentage contribution to total dose rate from EPRI's accident Classes 3a and 3b given an 1-in-15 years ILRT interval

$CLASS\_3a\_DOSE_{COMBINED\text{-}15}$ = combined internal and external events EPRI accident Class 3a dose rate given a 1-in-15 years ILRT interval

= $CLASS\_3a\_DOSE_{INTERNAL\text{-}15}$ + $CLASS\_3a\_DOSE_{EXTERNAL\text{-}15}$

$CLASS\_3b\_DOSE_{COMBINED\text{-}15}$ = combined internal and external events EPRI accident Class 3b dose rate given a 1-in-15 years ILRT interval

= $CLASS\_3b\_DOSE_{INTERNAL\text{-}15}$ + $CLASS\_3b\_DOSE_{EXTERNAL\text{-}15}$

$CLASS\_3a\_DOSE_{INTERNAL\text{-}15}$ = internal events EPRI accident Class 3a dose rate given a 1-in-15 years ILRT interval = 6.25×10^{-3} person-rem/ry [Table 2-10]

$CLASS\_3b\_DOSE_{INTERNAL\text{-}15}$ = internal events EPRI accident Class 3b dose rate given a 1-in-15 years ILRT interval = 2.19×10^{-3} person-rem/ry [Table 2-10]

$CLASS\_3a\_DOSE_{EXTERNAL\text{-}15}$ = external events EPRI accident Class 3a dose rate given a 1-in-15 years ILRT interval = 3.44×10^{-1} person-rem/ry [See for 1-in-15 years table above]

$CLASS\_3b\_DOSE_{EXTERNAL\ 15}$ = external events EPRI accident Class 3b dose rate given a 1-in-15 years ILRT interval = 1.20×10^{-1} person-rem/ry [See for 1-in-15 years table above]

$TOT\text{-}DOSE_{COMBINED\text{-}15}$ = Total combined internal and external events dose rate for all EPRI's Classes given a 1-in-15 years ILRT interval

= $TOT\text{-}DOSE_{INTERNAL\text{-}15}$ + $TOT\text{-}DOSE_{EXTERNAL\text{-}15}$

$TOT\text{-}DOSE_{INTERNAL\text{-}15}$ = Total internal events dose rate for all EPRI's Classes given a 1-in-15 years ILRT interval = 22.134 (person-rem/ry) [Table 2-10]

$TOT\text{-}DOSE_{EXTERNAL\text{-}15}$ = Total external events dose rate for all EPRI's Classes given a 1-in-15 years ILRT interval = 259.141 (person-rem/ry) [See for 1-in-10 years table above]

Therefore,

$$PER\_CHG_{COMBINED\text{-}15} = \left[\frac{(6.25 \times 10^{-3} + 3.44 \times 10^{-1}) + (2.19 \times 10^{-3} + 1.20 \times 10^{-1})}{22.134 + 259.141} \right] * 100$$

$$PER\_CHG_{COMBINED\text{-}15} = 0.1680\%$$

Based on the above results, the combined internal and external events changes from the 1-in-10 years to 1-in-15 years dose rate is as follows:

$$INCREASE_{COMBINED10\text{-}15} = \left[\frac{TOT\text{-}DOSE_{COMBINED\text{-}15} - TOT\text{-}DOSE_{COMBINED\text{-}10}}{TOT\text{-}DOSE_{COMBINED\text{-}10}} \right] * 100$$

Where:

$INCREASE_{COMBINED10\text{-}15}$ = combined internal and external events percent change from 1-in-10 years ILRT interval to 1-in-15 years ILRT interval

$TOT\text{-}DOSE_{COMBINED\text{-}15}$ = Total combined internal and external events dose rate for all EPRI's Classes given a 1-in-15 years ILRT interval

= $TOT\text{-}DOSE_{INTERNAL\text{-}15} + TOT\text{-}DOSE_{EXTERNAL\text{-}15}$

$TOT\text{-}DOSE_{COMBINED\text{-}10}$ = Total combined internal and external events dose rate for all EPRI's Classes given a 1-in-10 years ILRT interval

= $TOT\text{-}DOSE_{INTERNAL\text{-}10} + TOT\text{-}DOSE_{EXTERNAL\text{-}10}$

$TOT\text{-}DOSE_{INTERNAL\text{-}15}$ = Total internal events dose rate for all EPRI's Classes given a 1-in-15 years ILRT interval = 22.134 (person-rem/ry) [Table 2-10]

$TOT\text{-}DOSE_{EXTERNAL\text{-}15}$ = Total external events dose rate for all EPRI's Classes given a 1-in-10 years ILRT interval = 259.141 (person-rem/ry) [See for 1-in-10 years table above]

$TOT\text{-}DOSE_{INTERNAL\text{-}10}$ = Total internal events dose rate for all EPRI's Classes given a 1-in-15 years ILRT interval = 22.132 (person-rem/ry) [Table 2-10]

$TOT\text{-}DOSE_{EXTERNAL\text{-}10}$ = Total external events dose rate for all EPRI's Classes given a 1-in-10 years ILRT interval = 258.998 (person-rem/ry)[See for 1-in-10 years table above]

Therefore,

$$INCREASE_{COMBINED10\text{-}15} = \left[\frac{(22.134 + 259.141) - (22.132 + 258.998)}{(22.132 + 258.998)} \right] * 100$$

$$INCREASE_{COMBINED10\text{-}15} = 0.052\%$$

The above increase in risk on the total integrated plant risk for those accident sequences influenced by Type A testing, given the change from a 1-in-10 years test interval to a 1-in-15 years test interval, is found to be 0.052%. This value can be considered to be a negligible increase in risk.

Evaluate the External Events Hazard Risk Impact in Terms of LERF

This step, per the NEI Interim Guidance [4] calculates the change in the large early release frequency with extending the ILRT interval from 1-in-10 years to 1-in15-years.

The combined internal and external events affect on the LERF risk measure due to the proposed ILRT interval extension is calculated as follows:

$$)LERF_{COMBINED10\text{-}15} = CLASS\_3b_{COMBINED15} - CLASS\_3b_{COMBINED10}$$

Where:

$)LERF_{COMBINED10\text{-}15}$ = the combined internal and external events change in LERF from 1-in-10 years ILRT interval to 1-in-15 years ILRT interval

$CLASS\_3b_{COMBINED15}$ = the combined internal and external frequency of EPRI accident Class 3b given a 1-in-15 years ILRT Interval

= $CLASS\_3b_{INTERNAL\text{-}15} + CLASS\_3b_{EXTERNAL\text{-}15}$

$CLASS\_3b_{INTERNAL\text{-}15}$ = internal events frequency of EPRI accident Class 3b given a 1-in-15 years ILRT Interval = 5.90×10^{-9}/ry [Table 2-9]

$CLASS\_3b_{EXTERNAL\text{-}15}$ = external events frequency of EPRI accident Class 3b given a 1-in-15 years ILRT Interval = 3.24×10^{-7}/ry [See for 1-in-15 years table above]

$CLASS\_3b_{COMBINED10}$ = the combined internal and external frequency of EPRI accident Class 3b given a 1-in-10 years ILRT Interval

= $CLASS\_3b_{INTERNAL\text{-}10} + CLASS\_3b_{EXTERNAL\text{-}10}$

$CLASS\_3b_{INTERNAL\text{-}10}$ = internal events frequency of EPRI accident Class 3b given a 1-in-10 years ILRT Interval = 3.93×10^{-9}/ry [Table 2-9]

$CLASS\_3b_{EXTERNAL\text{-}10}$ = external events frequency of EPRI accident Class 3b given a 1-in-10 years ILRT Interval = 2.16×10^{-7}/ry [See for 1-in-10 years table above]

Therefore,

$$)LERF_{COMBINED10\text{-}15} = (5.90 \times 10^{-9} + 3.24 \times 10^{-7}) - (3.93 \times 10^{-9} + 2.16 \times 10^{-7})$$

$$)LERF_{COMBINED10\text{-}15} = 1.10 \times 10^{-7}/ry$$

The risk acceptance criteria of Regulatory Guide 1.174 as previously discussed in Section 7, Step 8 of this calculation, is used here to assess the ILRT interval extension. Regulatory Guide 1.174, "An Approach for Using PRA in Risk-Informed Decisions on Plant-Specific Changes to the Licensing Basis" [6], provides NRC recommendations for using risk information in support of applications requesting changes to the license basis of the plant.

The $)LERF_{COMBINED10\text{-}15}$ of 1.10×10^{-7}/ry from extending the Pilgrim Station ILRT frequency from 1-in-10 years to 1-in-15 years is within Region II of Regulatory Guide 1.174 acceptance guidelines. Therefore, per Regulatory Guide 1.174, since the calculated increase in LERF due to the proposed ILRT test interval

change is in the range of 10^{-7} to10^{-6} per reactor year, the risk assessment must also realistically show that the total LERF is less than 10^{-5}/yr.

From the Pilgrim Station internal events PSA [7] documentation, the Pilgrim Station LERF due to internal event accidents is 1.13×10^{-7}/ry. However, explicit information on LERF due to external events is not available from the Pilgrim Station IPEEE. Therefore, assuming a conservative LERF contribution that approximates 10% of CDF (note that the Pilgrim Station internal events LERF versus CDF relationship is approximately 1.76%), the Pilgrim Station LERF due to external events can be approximated by $0.10 \times 7.19 \times 10^{-5}$/ry = 7.19×10^{-6}/ry. Therefore, the total LERF for Pilgrim can be estimated at 1.13×10^{-7}/ry (internal events) + 7.19×10^{-6}/ry (external events) = 7.30×10^{-6}/ry. This value is than the Regulatory Guide 1.174 acceptance guideline of 10^{-5}/yr.

Evaluate the External Events Hazard Change in Conditional Containment Failure Probability

This step calculates the change in conditional containment failure probability (CCFP).

Similar to Section 2.4.9, Step 9 of this calculation, the change in CCFP tracts the impact of the ILRT on both early (LERF) and late radionuclide releases. Therefore, CCFP consists of all those accident sequences resulting in a radionuclide release other that the intact containment state for EPRI accident Class 1, and small failures state for EPRI accident Class 3a. In additional, the CCFP is conditional given a severe core damage accident. The change in CCFP is calculated by the following equation:

$$CCFP = \{1-([\text{Class 1 frequency} + \text{Class 3a frequency}]/CDF)\}*100, \%$$

For the combined internal and external events 1-in-10 years ILRT interval:

$$CCFP_{COMBINED\text{-}10} = \left\{ 1 - \left[\frac{CLASS\_1_{COMBINED\text{-}10} + CLASS\_3a_{COMBINED\text{-}10}}{CDF_{COMBINED}} \right] \right\} * 100\%$$

Where:

$CCFP_{COMBINED\text{-}10}$ = combined internal and external events conditional containment failure probability given 1-in-10 years ILRT interval

$CLASS\_1_{COMBINED\text{-}10}$ = combined internal and external events frequency of EPRI accident Class 1 given a 1-in-15 years ILRT interval
= $CLASS\_1_{INTERNAL\text{-}10} + CLASS\_1_{EXTERNAL\text{-}10}$

$CLASS\_3a_{COMBINED\text{-}10}$ = combined internal and external events frequency of EPRI accident Class 3a given a 1-in-10 years ILRT interval
= $CLASS\_3a_{INTERNAL\text{-}10} + CLASS\_3a_{EXTERNAL\text{-}10}$

$CLASS\_1_{INTERNAL\text{-}10}$ = internal events frequency of EPRI accident Class 1given a 1-in-10 years ILRT interval = 6.78×10^{-8}/ry [Table 2-9]

$CLASS\_1_{EXTERNAL\text{-}10}$ = external events frequency of EPRI accident Class 1given a 1-in-10 years ILRT interval = 7.45×10^{-6}/ry [See for 1-in-10 years table above]

$CLASS\_3a_{INTERNAL\text{-}10}$ = internal events frequency of EPRI accident Class 3a given a 1-in-10 years ILRT interval = 3.93×10^{-8}/ry [Table 2-9]

$CLASS\_3a_{EXTERNAL\text{-}10}$ = external events frequency of EPRI accident Class 3a given a 1-in-10 years ILRT interval = 2.16×10^{-6}/ry /ry [See for 1-in-10 years table above]

$CDF_{COMBINED}$ = Pilgrim Station combined internal events and external events CDF
= 6.41×10^{-6}/ry [Section 5, input#2] + 7.19×10^{-5}/ry [Section A5.2, input#5]
= 7.83×10^{-5}/ry

Therefore,

$$CCFP_{COMBINED\text{-}10} = \left\{ 1 - \left[\frac{(6.78 \times 10^{-8} + 7.45 \times 10^{-6}) + (3.93 \times 10^{-8} + 2.16 \times 10^{-6})}{7.83 \times 10^{-5}} \right] \right\} * 100\%$$

$CCFP_{COMBINED\text{-}10}$ = 87.59%

For the combined internal and external events 1-in-15 years ILRT interval:

$$CCFP_{COMBINED\text{-}15} = \left\{ 1 - \left[\frac{CLASS\_1_{COMBINED\text{-}15} + CLASS\_3a_{COMBINED\text{-}15}}{CDF_{COMBINED}} \right] \right\} * 100\%$$

Where:

$CCFP_{COMBINED\text{-}15}$ = combined internal and external events conditional containment failure probability given 1-in-15 years ILRT interval

$CLASS\_1_{COMBINED\text{-}15}$ = combined internal and external events frequency of EPRI accident Class 1 given a 1-in-15 years ILRT interval
= $CLASS\_1_{INTERNAL\text{-}15}$ + $CLASS\_1_{EXTERNAL\text{-}15}$

$CLASS\_3a_{COMBINED\text{-}15}$ = combined internal and external events frequency of EPRI accident Class 3a given a 1-in-15 years ILRT interval
= $CLASS\_3a_{INTERNAL\text{-}15}$ + $CLASS\_3a_{EXTERNAL\text{-}15}$

$CLASS\_1_{INTERNAL\text{-}15}$ = internal events frequency of EPRI accident Class 1given a 1-in-15 years ILRT interval = 4.61×10^{-8}/ry [Table 2-9]

$CLASS\_1_{EXTERNAL\text{-}15}$ = external events frequency of EPRI accident Class 1given a 1-in-15 years ILRT interval = 6.27×10^{-6}/ry [See for 1-in-15 years table above]

$CLASS\_3a_{INTERNAL\text{-}15}$ = internal events frequency of EPRI accident Class 3a given a 1-in-15 years ILRT interval = 5.90×10^{-8}/ry [Table 2-9]

$CLASS\_3a_{EXTERNAL\text{-}15}$ = external events frequency of EPRI accident Class 3a given a 1-in-15 years ILRT interval = 3.24×10^{-6}/ry /ry [See for 1-in-15 years table above]

$CDF_{COMBINED}$ = Pilgrim Station combined internal events and external events CDF
= 6.41×10^{-6}/ry [Section 5, input#2] + 7.19×10^{-5}/ry [Section A5.2, input#5]
= 7.83×10^{-5}/ry

Therefore,

$$CCFP_{COMBINED\text{-}15} = \left\{ 1 - \left[\frac{(4.61 \times 10^{-8} + 6.27 \times 10^{-6}) + (5.90 \times 10^{-8} + 3.24 \times 10^{-6})}{7.83 \times 10^{-5}} \right] \right\} * 100\%$$

$CCFP_{COMBINED\text{-}15}$ = 87.72%

Therefore, the change in the combined internal and external events conditional containment failure probability from 1-in-10 years to 1-in-15 years is:

$)CCFP_{COMBINED10\text{-}15} = CCFP_{COMBINED15} - CCFP_{COMBINED10}$

$)CCFP_{COMBINED10\text{-}15}$ = 87.72% - 87.59%

$)CCFP_{COMBINED10\text{-}15}$ = 0.13%

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

The effects of external hazard risk on ILRT risk are shown in Table A-8. The combined internal and external events effect on the ILRT risk is shown in Table A-9. This Table combines the results of Table 2-8, 2-9, and 2-10 with the results depicted in Table A-8.

A6.0 Conclusions

This appendix discusses the risk-implication associated with external hazards in support of the Pilgrim Station Integrated Leak Rate Testing (ILRT) interval extension risk assessment. The following conclusions are derived from this evaluation

1. The $)LERF_{COMBINED10\text{-}15}$ of 1.10×10^{-7}/ry from extending the Pilgrim Station ILRT frequency from 1-in-10 years to 1-in-15 years is slightly above the 10^{-7}/yr criterion of Region III, Very Small Change in Risk (Figure 1), of the acceptance guidelines in NRC Regulatory Guide 1.174 [6]. Consequently, consistent with Regulatory Guide 1.174, the total Pilgrim Station LERF from internal and external events was calculated at 7.30×10^{-6}/ry to demonstrate that LERF is acceptable. This is significantly less than the Regulatory Guide 1.174 acceptance guideline of 10^{-5}/yr. Therefore, increasing the ILRT interval at Pilgrim from the currently allowed 1-in-10 years to 1-in-15 years is non-risk significant from a risk perspective.

2. The combined internal and external events increase in risk on the total integrated plant risk for those accident sequences influenced by Type A testing, given the change from a 1-in-10 years test interval to a 1-in-15 years test interval, is found to be 0.052% (0.145 person-rem/ry). This value can be considered to be a negligible increase in risk.

3. The change in the combined internal and external events conditional containment failure probability from 1-in-10 years to 1-in-15 years is 0.13%. A change in CCFP of less than 1% is insignificant from a risk perspective

4.

Table A-1

Pilgrim Seismic Plant Damage States Classification [18]

Seismic Class	Class Description	Sub-class	Scenario Definition	Point Estimate	% Of Total CDF	Seismic Core Damage Sequences	SPDS	Internal PDS
I	Seismic-induced sequences where the RCS is not breached and the containment integrity is not challenged prior to core melt. RCS inventory boil-off is through the SRVs to the suppression pool.		Seismic surrogate element.	1.63E-06	3.09%		SPDS-1	NA
		A	Early core melt at high RPV pressure with low-pressure systems at RCS depressurization (or at vessel breach). Torus is subcooled, as RHR is available.	2.61E-05	49.50%	SITQU	SPDS-2	PDS-14
II	Seismic-induced accident sequences in which containment decay heat removal systems are not available and coolant recirculation to the torus overpressurize the containment to failure or venting. The torus is saturated.	A	Accident sequences involving loss of containment heat removal with the RPV initially intact. Very late core melt at high RPV pressure induced post high containment pressure.	1.74E-05	32.88%	SITW	SPDS-3	PDS-13
		B	Accident sequences involving loss of containment heat removal with the RPV breached. Very late core melt induced post high containment pressure.	5.28E-09	0.01%	SILL1, SORV1, SORV3	SPDS-4	PDS-6

Table A-1

Seismic Plant Damage States Classification (continued) [18]

Seismic Class	Class Description	Sub-class	Scenario Definition	Point Estimate	% Of Total CDF	Seismic Core Damage Sequences	SPDS	Internal PDS
III	Seismic-induced LOCA initiated sequences in which RCS pressure and leakage rates associated with large break LOCA's with the occurrence of early core melt. Containment integrity is maintained prior to core damage.	A	Accident sequences initiated or resulting in small or medium LOCAs for which the RPV cannot be depressurized prior to core damage occurring.	2.88E-06	5.46%	SISL2, SIML3	SPDS-5	PDS-3 (revised)
		B	Accident sequences initiated or resulting in medium or large LOCAs for which the RPV is at low pressure.	0.00E+00	0.00%	SIML2, SILL2	SPDS-6	PDS-9
		C	Accident sequences initiated or resulting in large LOCAs or vessel rupture for which effective injection is beyond core standby cooling systems capabilities and the Vapor Suppression System is inadequate, challenging containment integrity.	3.90E-06	7.39%	SISL1, SIML1, SIVR, SORV2, SORV4	SPD-7	PDS_45 (revised)
IV	Seismic-induced ATWS sequence at high RPV pressure and rapid containment pressurization. RCS leakage rates associated with boiloff of coolant through the cycling of SRVs/SV with early core melt subsequent to containment overpressure failure.	A	Accident sequences involving failure of adequate shutdown reactivity with the RPV initially intact. Core damage induced post high containment pressure.	5.91E-07	1.12%	S1ATWS	SPDS-8	PDS-45
V	Seismic-induced LOCA outside containment and failure of coolant injection, resulting in early core melting.	A	Unisolated LOCA outside containment with core melt at high/low RPV pressure.	2.90E-07	0.55%	SIAOUT1, SIAOUT2, SIISL1, SIISL2	SPDS-9	PDS-48

Table A-2

Pilgrim Seismic Release Bins Frequencies [18]

Seismic Release Bin	Seismic Release Bin Description	Frequency (/year)	Percent of CDF
L2CONTFL	Containment Bypass Failure	4.66×10^{-6}	8.82%
L2LSISOL	Containment Isolation Failure	1.63×10^{-7}	0.31%
L2QUSTRX	Containment Structural Failure	2.47×10^{-6}	4.69%
L2SCFE	Early Containment Release	3.62×10^{-6}	6.85%
L2SCFL	Late Containment Release	3.11×10^{-5}	58.83%
NCF	No Containment Failure	9.19×10^{-6}	17.40%
SURR-CFE	Surrogate Early Release	1.90×10^{-7}	0.36%
SURR-CFL	Surrogate Late Release	9.09×10^{-7}	1.72%
SURR-NCF	Surrogate No Release	5.33×10^{-7}	1.01%
Total		5.28×10^{-5}	1.00

Table A-3

Summary of Seismic Release Bins Allocated to Classes 2, 7 and 8 of the EPRI Classification Scheme

EPRI Severe Accident Type	Seismic Release Bin	Definition	Frequency (/year)
2	L2LSISOL	Vessel breach occurs with a subsequent failure to isolate containment.	1.63×10^{-7}
7a	L2QUSTRX, L2SCFE, SURR-CFE	Vessel breach occurs and both the containment and the drywell have failed either before or at the time of vessel breach.	6.28×10^{-6}
7b	NA		
7c	L2SCFL, SURR-CFL	Vessel breach occurs, however, the containment does not fail until the late time period.	3.20×10^{-5}
7d	NA		
8	L2CONTFL	Vessel breach occurs with containment bypassed.	4.66×10^{-6}

Table A-4
Pilgrim Fire PRA Dominant Core Damage Sequences [18]

Sub-Area	Description	Frequency (/yr)
1E	Reactor Building West, El. 21	8.25×10^{-7}
2B	Turbine Building Heater Bay	2.74×10^{-6}
3A	Train "B" RBCCW/TBCCW Pump and Heat Exchanger Room	1.31×10^{-6}
4A	Train "A" RBCCW/TBCCW Pump and Heat Exchanger Room	2.95×10^{-7}
6	Control Room	8.90×10^{-7}
7	Cable Spreading Room	7.85×10^{-7}
9	Vital Motor Generator Set Room	2.38×10^{-6}
12	Train "A" Switchgear Room	2.30×10^{-6}
13	Train "B" Switchgear Room	6.85×10^{-6}
26	Main Transformer	7.60×10^{-7}
Total		1.91×10^{-5}

Table A-5

Pilgrim Fire Events Plant Damage States Classification [18]

Fire Class	Class Description	Sub-class	Scenario Definition	Point Estimate	% Of Total CDF	FPDS	Internal PDS
I	Transient-initiated sequences where the RCS is not breached and the containment integrity is not challenged prior to core melt. RCS inventory boil-off is through the SRVs to the suppression pool.	A	Early core melt at high RPV pressure with low-pressure systems at RCS depressurization (or at vessel breach). Torus is subcooled, as RHR is available.	3.09×10^{-7}	1.62%	FPDS-1	14
		B	Early core melt at low RPV pressure and failure of low-pressure systems. RHR is available to mitigate containment pressure and provide torus cooling.	9.51×10^{-9}	0.05%	FPDS-2	25
		F	Station blackout sequences involving early core melt at low RPV pressure from either two SORVs or failure of HPC/RCIC and one SORV. All accident-mitigating functions recoverable when ac power is restored.	1.24×10^{-10}	0.00%	FPDS-3	32
		G	Station blackout sequences involving late core melt at high RPV pressure from battery depletion. All accident-mitigating functions are recoverable when ac power is restored.	7.76×10^{-7}	4.06%	FPDS-4	29
		H	Station blackout sequences involving late core melt at low RPV pressure from either one stuck-open SRV or long-term failure of HPCI/RCIC and subsequent failure to depressurize the primary system. All accident-mitigating functions are recoverable when offsite power is restored.	8.42×10^{-9}	0.04%	FPDS-5	31
		K	Similar to IA, except that containment venting is not available.	1.69×10^{-9}	0.01%	FPDS-6	15
		M	Similar to IB, except that containment venting is not available.	5.71×10^{-9}	0.03%	FPDS-7	26

Table A-5

Fire Events Plant Damage States Classification (continued) [18]

Fire Class	Class Description	Sub-class	Scenario Definition	Point Estimate	% Of Total CDF	FPDS	Internal PDS
II	Containment decay heat removal systems are not available and coolant recirculation to the torus overpressurizes the containment to failure or venting. The torus is saturated.	A	Accident sequences involving loss of containment heat removal with the RPV initially intact. Very late core melt at high RPV pressure induced post high containment pressure.	4.81×10^{-6}	25.11%	FPDS-8	13
		B	Accident sequences involving loss of containment heat removal with the RPV breached. Very late core melt induced post high containment pressure.	7.05×10^{-6}	36.84%	FPDS-9	19
		C	Similar to IIA except that containment vent operates. Late core damage occurs on loss of RPV makeup after vent initiation. Torus is saturated but remains intact.	8.06×10^{-8}	0.42%	FPDS-10	12
		D	Similar to IIB except that containment vent operates. Late core damage occurs on loss of RPV makeup after vent initiation. Torus is saturated but remains intact.	6.07×10^{-6}	31.72%	FPDS-11	18
		E	Accident sequences initiated or resulting in medium o r large LOCAs for which the RPV is at low pressure and low-pressure injection is available.	1.76×10^{-8}	0.09%	FPDS-12	24

Table A-6

Pilgrim Fire Release Bins Frequencies [18]

Fire Release Bin	Fire Release Bin Description	Frequency (/yr)	Percent of CDF
FCAPB-1	[CD, No VB, No CF, No CCI] Core damage occurs (CD), but the recovery of RPV injection in time prevents vessel beach (No VB). Therefore, containment integrity is not challenged (No CF) and core-concrete interactions are precluded (No CCI). However, the potential exists for some in-vessel release to the environment due to containment design leakage.	1.06×10^{-7}	0.55
FCAPB-2	[CD, VB, No CF, No CCI] Core damage occurs (CD) followed by vessel breach (VB). The containment does not fail structurally and is not vented (No CF). Ex-vessel releases are recovered, therefore precluding the occurrence of core-concrete interactions (No CCI). Although the containment does not fail, vessel breach did occur, therefore the potential exists for some in- and ex-vessel releases to the environment due to containment design leakage. RPV pressure is not important because, even though high pressure induced severe accident phenomena (such as direct containment heating [DCH]) occurred, it did not fail containment.	1.23×10^{-8}	0.06
FCAPB-3	[CD, VB, No CF, CCI] Core damage occurs (CD) followed by vessel breach (VB). The containment does not fail structurally and is not vented (No CF). However, ex-vessel releases are not recovered in time, and therefore core-concrete interactions occur (CCI). RPV pressure is not important because, high pressure induced severe accident phenomena even if it occurred does not significantly affect the source term as the containment does not fail nor is the vent limit reached.	1.55×10^{-9}	0.01
FCAPB-4	[CD, VB, Early CF, WW, RPV pressure >200 psig at VB, No CCI] Core damage (CD) occurs followed by vessel breach (VB). The containment fails either before core damage, during core damage or at vessel breach (Early CF). The containment failure occurs in the torus (WW), above the water level. RPV pressure is greater than 200 psig at the time of vessel breach (this implies that high pressure induced severe accident phenomena [DCH] is possible). There are no core concrete interactions (No CCI) due to the present of an overlying pool of water.	3.16×10^{-9}	0.02

CD = core damage VB = vessel breach CF = containment failure
DW = drywell WW = torus RPV = reactor pressure vessel
CCI = core-concrete interactions

Table A-6

Pilgrim Fire Release Bins Frequencies [18] (Continued)

Fire Release Bin	Fire Release Bin Description	Frequency (/yr)	Percent of CDF
FCAPB-5	[CD, VB, Early CF, WW, RPV pressure <200 psig at VB, No CCI] Core damage (CD) occurs followed by vessel breach (VB). The containment fails either before core damage, during core damage or at vessel breach (Early CF). The containment failure occurs in the torus (WW), above the water level. RPV pressure is less than 200 psig at the time of vessel breach; thus, precluding high pressure induced severe accident phenomena. There are no core concrete interactions (No CCI) due to the present of an overlying pool of water.	3.33×10^{-9}	0.02
FCAPB-6	[CD, VB, Early CF, WW, RPV pressure >200 psig at VB, CCI] Core damage (CD) occurs followed by vessel breach (VB). The containment fails either before core damage, during core damage or at vessel breach (Early CF). The containment failure occurs in the torus (WW), above the water level. RPV pressure is greater than 200 psig at the time of vessel breach (this implies that high pressure induced severe accident phenomena [DCH] is possible). Following containment failure, core-concrete interactions occurs (CCI).	1.82×10^{-8}	0.10
FCAPB-7	[CD, VB, Early CF, WW, RPV pressure <200 psig at VB, CCI]Core damage (CD) occurs followed by vessel breach (VB). The containment fails either before core damage, during core damage or at vessel breach (Early CF). The containment failure occurs in the torus (WW), above the water level. RPV pressure is less than 200 psig at the time of vessel breach; thus, precluding high pressure induced severe accident phenomena. Following containment failure, core-concrete interactions occurs (CCI).	2.81×10^{-8}	0.15
FCAPB-8	[CD, VB, Early CF, DW, RPV pressure >200 psig at VB, No CCI] Core damage (CD) occurs followed by vessel breach (VB). The containment fails either before core damage, during core damage or at vessel breach (Early CF). The containment failure occurs in the drywell or below the torus water line (DW). RPV pressure is greater than 200 psig at the time of vessel breach (this implies that high pressure induced severe accident phenomena [DCH] is possible). There are no core concrete interactions (No CCI) due to the present of an overlying pool of water.	2.97×10^{-8}	0.16

CD = core damage VB = vessel breach CF = containment failure
DW = drywell WW = torus RPV = reactor pressure vessel
CCI = core-concrete interactions

Table A-6

Pilgrim Fire Release Bins Frequencies [18] (Continued)

Fire Release Bin	Fire Release Bin Description	Frequency (/yr)	Percent of CDF
FCAPB-9	[CD, VB, Early CF, DW, RPV pressure <200 psig at VB, No CCI] Core damage (CD) occurs followed by vessel breach (VB). The containment fails either before core damage, during core damage or at vessel breach (Early CF). The containment failure occurs in the drywell or below the torus water line (DW). RPV pressure is less than 200 psig at the time of vessel breach (this implies that high pressure induced severe accident phenomena is precluded). There are no core concrete interactions (No CCI) due to the present of an overlying pool of water.	1.72×10^{-8}	0.09
FCAPB-10	[CD, VB, Early CF, DW, RPV pressure >200 psig at VB, CCI] Core damage (CD) occurs followed by vessel breach (VB). The containment fails either before core damage, during core damage or at vessel breach (Early CF). The containment failure occurs in the drywell or below the torus water line (DW). RPV pressure is greater than 200 psig at the time of vessel breach (this implies that high pressure induced severe accident phenomena [DCH] is possible). Following containment failure, core-concrete interactions occurs (CCI).	1.89×10^{-7}	0.99
FCAPB-11	[CD, VB, Early CF, DW, RPV pressure <200 psig at VB, CCI] Core damage (CD) occurs followed by vessel breach (VB). The containment fails either before core damage, during core damage or at vessel breach (Early CF). The containment failure occurs in the drywell or below the torus water line (DW). RPV pressure is less than 200 psig at the time of vessel breach; thus, precluding high pressure induced severe accident phenomena. Following containment failure, core-concrete interactions occurs (CCI).	1.47×10^{-7}	0.77
FCAPB-12	[CD, VB, Late CF, WW, No CCI] Core damage (CD) occurs followed by vessel breach (VB). The containment fails late due to a loss of containment heat removal (Late CF). The containment failure occurs in the torus (WW), above the water level. RPV pressure is not important because if a high-pressure severe accident phenomena (such as DCH) occurred, it did not fail containment upon its occurrence. There are no core concrete interactions (No CCI) due to the present of an overlying pool of water.	7.53×10^{-6}	39.35

CD = core damage VB = vessel breach CF = containment failure
DW = drywell WW = torus RPV = reactor pressure vessel
CCI = core-concrete interactions

Table A-6

Pilgrim Fire Release Bins Frequencies [18] (Continued)

Fire Release Bin	Fire Release Bin Description	Frequency (/yr)	Percent of CDF
FCAPB-13	[CD, VB, Late CF, WW, CCI] Core damage (CD) occurs followed by vessel breach (VB). The containment fails late (late CF) due to core-concrete interactions (CCI) after vessel breach. The containment failure occurs in the torus (WW), above the water level. RPV pressure is not important because, although a high-pressure severe accident phenomena (such as DCH) occurred, it did not fail containment.	1.60×10^{-9}	0.01
FCAPB-14	[CD, VB, Late CF, DW, No CCI] rs followed by vessel breach (VB). The containment fails late due to a loss of containment heat removal (Late CF). The containment failure occurs in either the drywell or below the torus water level (DW). RPV pressure is not important, because the occurrence of a high-pressure severe accident phenomenon did not fail containment. There are no core concrete interactions (No CCI) due to the present of an overlying pool of water.	3.90×10^{-6}	20.38
FCAPB-15	[CD, VB, Late CF, DW, CCI] Core damage (CD) occurs followed by vessel breach (VB). The containment fails late (late CF) due to core-concrete interactions (CCI) after vessel breach. The containment failure occurs in either the drywell or below the torus water level (DW). RPV pressure is not important because, if a high-pressure severe accident phenomenon occurred, it did not fail containment upon its occurrence.	7.15×10^{-6}	37.36

CD = core damage VB = vessel breach CF = containment failure
DW = drywell WW = torus RPV = reactor pressure vessel
CCI = core-concrete interactions

Table A-7

Summary of Fire Release Bins Allocated to Classes 2, 7 and 8 of the EPRI Classification Scheme

EPRI Severe Accident Type	Fire Release Bin	Definition	Frequency (/year)
2	NA[14]		
7a	FCAPB-8, FCAPB-9, FCAPB-10, FCAPB-11	Failure Induced by Phenomena (Early Drywell Failures)	3.83×10^{-7}
7b	FCAPB-4, FCAPB-5, FCAPB-6, FCAPB-7	Failure Induced by Phenomena (Early Torus Failures)	5.28×10^{-8}
7c	FCAPB-14, FCAPB-15	Failure Induced by Phenomena (Late Drywell Failures)	1.11×10^{-5}
7d	FCAPB-12, FCAPB-13	Failure Induced by Phenomena (Late Torus Failures)	7.53×10^{-6}
8	NA	Bypass (ATWS, ISLOCA)	

[14] Value of 6.9×10^{-6} from internal events is used.

Table A-8

Effect of External Events Hazard Risk on Pilgrim ILRT Risk Assessment

	Base Case 3 Years			Extend to 10 Years			Extend to 15 Years		
EPRI Class	CDF (Per Ry)	Per-Rem	Per-Rem (Per Ry)	CDF (Per Ry)	Per-Rem	Per-Rem (Per Ry)	CDF (Per Ry)	Per-Rem	Per-Rem (Per Ry)
1	9.12×10^{-6}	1.06×10^{4}	9.67×10^{-2}	7.45×10^{-6}	1.06×10^{4}	7.91×10^{-2}	6.27×10^{-6}	1.06×10^{4}	6.65×10^{-2}
2	1.63×10^{-7}	4.53×10^{6}	7.37×10^{-1}	1.63×10^{-7}	4.53×10^{6}	7.37×10^{-1}	1.63×10^{-7}	4.53×10^{6}	7.37×10^{-1}
3a	6.49×10^{-7}	1.06×10^{5}	6.88×10^{-2}	2.16×10^{-6}	1.06×10^{5}	2.29×10^{-1}	3.24×10^{-6}	1.06×10^{5}	3.44×10^{-1}
3b	6.49×10^{-8}	3.71×10^{5}	2.41×10^{-2}	2.16×10^{-7}	3.71×10^{5}	8.02×10^{-2}	3.24×10^{-7}	3.71×10^{5}	1.20×10^{-1}
4	0.00	N/A	0.00	0.00	N/A	0.00	0.00	N/A	0.00
5	0.00	N/A	0.00	0.00	N/A	0.00	0.00	N/A	0.00
6	0.00	N/A	0.00	0.00	N/A	0.00	0.00	N/A	0.00
7a	6.66×10^{-6}	4.53×10^{6}	3.02×10^{1}	6.66×10^{-6}	4.53×10^{6}	1.90×10^{1}	6.66×10^{-6}	4.53×10^{6}	1.90×10^{1}
7b	5.28×10^{-8}	1.82×10^{6}	9.61×10^{-2}	5.28×10^{-8}	1.82×10^{6}	9.61×10^{-2}	5.28×10^{-8}	1.82×10^{6}	9.61×10^{-2}
7c	4.30×10^{-5}	4.55×10^{6}	1.96×10^{2}	4.30×10^{-5}	4.55×10^{6}	1.91×10^{2}	4.30×10^{-5}	4.55×10^{6}	1.91×10^{2}
7d	7.53×10^{-6}	7.35×10^{5}	5.54×10^{-0}	7.53×10^{-6}	7.35×10^{5}	5.54×10^{0}	7.53×10^{-6}	7.35×10^{5}	5.54×10^{0}
8	4.66×10^{-6}	5.66×10^{6}	2.63×10^{1}	4.66×10^{-6}	5.66×10^{6}	2.64×10^{1}	4.66×10^{-6}	5.66×10^{6}	2.64×10^{1}
Total	7.19×10^{-5}		258.800	7.19×10^{-5}		258.998	7.19×10^{-5}		259.141
ILRT Dose Rate from 3a and 3b			9.28×10^{-2}			3.09×10^{-1}			4.64×10^{-1}
% Of Total			0.0359%			0.1195%			0.1791%
Delta Dose Rate from 3a and 3b (10 to 15 yr)									0.155
LERF from 3b			6.50×10^{-8}			2.16×10^{-7}			3.24×10^{-7}
Delta LERF (10 to 15 yr)									1.08×10^{-7}/ry
CCFP %			86.4%			86.6%			86.8%
Delta CCFP % (10 to 15 yr)									0.2 %

Table A-9

Effect of Internal and External Events Risk on Pilgrim ILRT Risk Assessment

	Base Case 3 Years			Extend to 10 Years			Extend to 15 Years		
EPRI Class	CDF (Per Ry)	Per-Rem	Per-Rem (Per Ry)	CDF (Per Ry)	Per-Rem	Per-Rem (Per Ry)	CDF (Per Ry)	Per-Rem	Per-Rem (Per Ry)
1	9.22×10^{-6}	1.06×10^{4}	9.78×10^{-2}	7.53×10^{-6}	1.06×10^{4}	7.98×10^{-2}	6.32×10^{-6}	1.06×10^{4}	6.70×10^{-2}
2	1.63×10^{-7}	4.53×10^{6}	7.38×10^{-1}	1.63×10^{-7}	4.53×10^{6}	7.38×10^{-1}	1.63×10^{-7}	4.53×10^{6}	7.38×10^{-1}
3a	6.60×10^{-7}	1.06×10^{5}	7.00×10^{-2}	2.20×10^{-6}	1.06×10^{5}	2.33×10^{-1}	3.30×10^{-6}	1.06×10^{5}	3.50×10^{-1}
3b	6.60×10^{-8}	3.71×10^{5}	2.45×10^{-2}	2.20×10^{-7}	3.71×10^{5}	8.17×10^{-2}	3.30×10^{-7}	3.71×10^{5}	1.22×10^{-1}
4	0.00	N/A	0.00	0.00	N/A	0.00	0.00	N/A	0.00
5	0.00	N/A	0.00	0.00	N/A	0.00	0.00	N/A	0.00
6	0.00	N/A	0.00	0.00	N/A	0.00	0.00	N/A	0.00
7a	6.82×10^{-6}	4.53×10^{6}	3.09×10^{1}	6.82×10^{-6}	4.53×10^{6}	3.09×10^{1}	6.82×10^{-6}	4.53×10^{6}	3.09×10^{1}
7b	7.47×10^{-8}	1.82×10^{6}	1.36×10^{-1}	7.47×10^{-8}	1.82×10^{6}	1.36×10^{-1}	7.47×10^{-8}	1.82×10^{6}	1.36×10^{-1}
7c	4.74×10^{-5}	4.55×10^{6}	2.16×10^{2}	4.74×10^{-5}	4.55×10^{6}	2.16×10^{2}	4.74×10^{-5}	4.55×10^{6}	2.16×10^{2}
7d	9.23×10^{-6}	7.35×10^{5}	6.79×10^{0}	9.23×10^{-6}	7.35×10^{5}	6.79×10^{0}	9.23×10^{-6}	7.35×10^{5}	6.79×10^{0}
8	4.69×10^{-6}	5.66×10^{6}	2.66×10^{1}	4.69×10^{-6}	5.66×10^{6}	2.66×10^{1}	4.69×10^{-6}	5.66×10^{6}	2.66×10^{1}
Total	7.83×10^{-5}		280.956	7.83×10^{-5}		281.159	7.83×10^{-5}		281.304
ILRT Dose Rate from 3a and 3b			9.45×10^{-2}			3.15×10^{-1}			4.72×10^{-1}
% Of Total			0.0336%			0.1120%			0.1680%
Delta Dose Rate from 3a and 3b (10 to 15 yr)									0.157
LERF from 3b			6.60×10^{-8}			2.20×10^{-7}			3.30×10^{-7}
Delta LERF (10 to 15 yr)									1.10×10^{-7}
CCFP %			87.38%			87.59%			87.72%
Delta CCFP % (10 to 15 yr)									0.13%

Appendix B

Risk Impact of Containment Liner Corrosion During an Extension of the ILRT Interval

Table of Contents

Tables

Figures

B1.0 Introduction

Inspections of reinforced and steel containments at some facilities (e.g., North Anna, Brunswick D.C. Cook, and Oyster Creek) have indicated degradation from the inaccessible side of the steel shell and liner of primary containments. The major inaccessible areas of the Mark I containment are the vertical portion of the drywell shell and part of the shell located between the drywell floor and the basemat. As a result of these inaccessible areas, a potential increase in risk due to liner leakage, caused by age-related degradation mechanisms may occur when extending the current 1-in-10 years to 1-in-15 years Type A Integrated Leak Rate Testing (ILRT) interval.

Therefore, this appendix evaluates the likelihood and risk-implication associated with containment liner corrosion going undetected in visual examinations during the proposed extension of the ILRT interval.

B2.0 Method of Analysis

The analysis utilizes the referenced Calvert Cliffs Nuclear Power Plant assessment [20] to estimate the risk impact from containment liner corrosion during an extension of the ILRT interval.

Consistent with the Calvert Cliffs analysis, the following issues are addressed:

- Differences between the containment basemat and the drywell and torus liner
- The historical drywell/torus steel shell 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 method of analysis determines the total likelihood of non-detected containment leakage given a change in the likelihood that a flaw exists (i.e., increase in flaw likelihood due to the ILRT extension), that the flaw is not detected and that flaw results in a breach.

Consistent with Calvert Cliffs analysis [20], the following six steps are performed:

1) Determine the historical liner flaw likelihood.
2) Determine aged adjusted liner flaw likelihood.
3) Determine the increase in flaw likelihood between 3, 10 and 15 years.
4) Determine the likelihood of containment breach given liner flaw.
5) Determine the visual inspection detection failure.
6) Determine the likelihood of non-detected containment leakage.

In additions to these steps, the following three additional steps are added to evaluate the risk-implication of containment liner corrosion:

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

8) Evaluate the risk impact in terms of LERF.

9) Evaluate the change in conditional containment failure probability.

B3.0 Assumptions

1) Consistent with the Calvert Cliffs methodology [20], a half failure is assumed for basemat concealed liner corrosion due to the lack of identified failures.

2) Consistent with the Calvert Cliffs methodology [20], the leakage potential via the drywell floor (due to crack formation) is considered less likely than other sections of the containment structure.

3) Consistent with the Calvert Cliffs methodology [20], the likelihood of the containment atmosphere reaching the outside atmosphere given a liner flaw exists was estimated as a function of the pressure inside the containment.

4) Consistent with the Calvert Cliffs methodology [20], the containment liner flaw likelihood doubles every five years. This is based solely on judgment and is included in this analysis to address the increase likelihood of corrosion as the containment liner ages.

5) Consistent with the Calvert Cliffs methodology [20], the probability of a concurrent containment breach given a flaw in the containment liner is depicted as an exponential function.

6) Consistent with the Calvert Cliffs methodology [20], a 0.05 (5%) visual inspection detection failure likelihood given the flaw is visible and a total detection failure likelihood of 0.10 (10%) is used[15].

7) Consistent with the Calvert Cliffs methodology [20], 1.0 (100%) visual inspection detection failure likelihood given the flaw is located in an inaccessible area of either the drywell or torus.

8) Consistent with the Calvert Cliffs methodology [20], all non-detectable containment failures are considered to result in large early releases.

B4.0 Input

1) The containment liner failure rate is based on two industry events:

   1. On September 22, 1999, North Anna Unit 2 experienced through-wall corrosion of the metal liner. The corrosion appeared to have been initiated from a piece of lumber imbedded in the concrete behind the liner plate.

   2. On April 27, 1999, inspection at Brunswick 2 discovered two through-wall holes and pitting in the drywell shell. The through-wall condition was believed to have originated from the coated (visible) side.

[15] Note: to date, all liner corrosion events have been detected through visual inspection.

2) The number of steel-lined containments is 70 [20].

3) The exposure time in detecting a containment flaw is 5.5 years. This is consistent with the Calvert Cliffs methodology [20] and reflects the time period since 10CFR 50.55a starting requiring visual inspection. This is deemed conservative, since the exposure time period is bounding as no additional failures have been identified in the nuclear industry since March 2002 and no failures were identified prior to September 1996 (the date when 10CFR 50.55a was implemented).

4) Consistent with the Calvert Cliffs methodology [20], leakage through the drywell floor is 10 times less likely than through other sections of the containment structure.

5) The probability of a concurrent containment breach given a flaw in the containment liner is depicted as an exponential function. This curve is used to interpolate the containment failure probability at the pressure at which the ILRT is to be performed for the accessible and inaccessible areas of containment. Consistent with the Calvert Cliffs methodology, the lower bound limit was assigned a failure probability of 0.1% at a pressure of 20 psia and the upper bound was assigned a failure probability of 100% at the ultimate containment failure pressure of 113 psia [7].

B5.0 Steel Shell Corrosion Analysis

Step 1B - Determine the Historical Liner Flaw Likelihood.

This step calculates historical liner flaw likelihood consistent wit the Calvert Cliffs mythology [20]. This value, for Pilgrim's consists of the accessible potion of the drywell and torus, the inaccessible portion of the drywell and submergence area of the torus, and the inaccessible area of the drywell floor.

The accessible portion of the drywell and torus liner flaw likelihood is determined as follows:

$$AHLF_{DT} = NFAIL_a \quad / \quad (NPLANTS * TEXPO)$$

The inaccessible portion of the drywell and submergence area of the torus liner flaw likelihood is determined as follows:

$$IAHLF_{DT} = NFAIL_a \quad / \quad (NPLANTS * TEXPO)$$

The inaccessible area of the drywell floor

$$IAHLF_{DF} = NFAIL_{ia} \quad / \quad (NPLANTS * TEXPO)$$

Where:

$AHLF_{DT}$	=	accessible portion of the drywell and torus liner flaw			
$IAHLF_{DT}$	=	inaccessible portion of the drywell and submergence area of the torus liner flaw likelihood			
$IAHLF_{DF}$	=	inaccessible area of the drywell floor liner flaw			
$NFAIL_a$	=	number of industry events due to liner corrosion	=	2	[Section B4.0, Input #1]
$NFAIL_{ia}$	=	number of industry events due basemat corrosion	=	0.5	[Section B3.0, Input #1]
NPLANTS	=	number of steel-lined containments = 70			[Section B4.0, Input #2]
TEXPO	=	time exposure since issuing of 10CFR50.55a	=	5.5 years	[Section B4.0, Input #3]

Therefore,

$$AHLF_{DT} = 2 / (70 * 5.5) = 5.19 \times 10^{-3}/yr$$

$$IAHLF_{DT} = 2 / (70 * 5.5) = 5.19 \times 10^{-3}/yr$$

$$IAHLF_{DF} = 0.5 / (70 * 5.5) = 1.30 \times 10^{-3}/yr$$

The above results are documented in Table B-4.

Step 2B - Determine Aged Adjusted Liner Flaw Likelihood.

Per the Calvert Cliffs methodology [20], the aged adjustment liner flaw likelihood is calculated for a 15-year interval given that the failure rate doubles every 5 years (Section B3.0, assumption #4) or increases 14.9 % per year. In addition, the average for the 5th to 10th year was set to the historical failure calculated in Step 1B.

The results, based on an iterative process that satisfies the above conditions are presented in Table B-1.

Step 3B - Determine the increase in flaw likelihood between 3, 10 and 15 years[16].

This step calculates the increase in flaw likelihood at 3-in-10 years interval (or 1-in-3 years), 1-in-10 years interval, and 1-in-15 years interval, per the Calvert Cliffs methodology [20]. The results of Step 2B are use to generate these values as follows:

Accessible portion of the drywell and torus,

$$ADTFLAW_{3\text{-}10} = \sum_{i=1,3} ADTF_{RATEii}$$

$$ADTFLAW_{1\text{-}10} = \sum_{i=1,10} ADTF_{RATEii}$$

$$ADTFLAW_{1\text{-}15} = \sum_{i=1,15} ADTF_{RATEii}$$

Inaccessible portion of the drywell and submergence area of the torus,

$$IDTFLAW_{3\text{-}10} = \sum_{i=1,3} IDTF_{RATEii}$$

$$IDTFLAW_{1\text{-}10} = \sum_{i=1,10} IDTF_{RATEii}$$

$$IDTFLAW_{1\text{-}15} = \sum_{i=1,15} IDTF_{RATEii}$$

[16] Note: 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-in-10 years, 1-in-10 years, and 1-in-15 years intervals consistent with the evaluation in this calculation, and then the delta-LERF values are determined from there.

Inaccessible area of the drywell floor

$$DFFLAW_{3\text{-}10} = \sum_{i=1,3} DFF_{RATEii}$$

$$DFFLAW_{1\text{-}10} = \sum_{i=1,10} DFF_{RATEii}$$

$$DFFLAW_{1\text{-}15} = \sum_{i=1,15} DFF_{RATEii}$$

Where:

$ADTFLAW_{3\text{-}10}$ = increase in flaw likelihood at 3-in-10 years test interval given accessible portion of the drywell and torus

$ADTFLAW_{1\text{-}10}$ = increase in flaw likelihood at 1-in-10 years test interval given accessible portion of the drywell and torus

$ADTFLAW_{1\text{-}15}$ = increase in flaw likelihood at 1-in-15 years test interval given accessible portion of the drywell and torus

$IDTFLAW_{3\text{-}10}$ = increase in flaw likelihood at 3-in-10 years test interval given inaccessible portion of the drywell and submergence area of the torus

$IDTFLAW_{1\text{-}10}$ = increase in flaw likelihood at 1-in-10 years test interval given inaccessible portion of the drywell and submergence area of the torus

$IDTFLAW_{1\text{-}15}$ = increase in flaw likelihood at 1-in-15 years test interval given inaccessible portion of the drywell and submergence area of the torus

$DFFLAW_{3\text{-}10}$ = increase in flaw likelihood at 3-in-10 years test interval given inaccessible area of the drywell floor

$DFFLAW_{1\text{-}10}$ = increase in flaw likelihood at 1-in-10 years test interval given inaccessible area of the drywell floor

$DFFLAW_{1\text{-}15}$ = increase in flaw likelihood at 1-in-15 years test interval given inaccessible area of the drywell floor

$ADTF_{RATEii}$ = aged adjusted liner flaw likelihood, given accessible portion of the drywell and torus (Table B-1)

$IDTF_{RATEii}$ = aged adjusted liner flaw likelihood, given inaccessible portion of the drywell and submergence area of the torus (Table B-1)

DFF_{RATEii} = aged adjusted liner flaw likelihood, given inaccessible area of the drywell floor (Table B-1)

Therefore,

$ADTFLAW_{3\text{-}10}$	=	0.71%,	$ADTFLAW_{1\text{-}10}$	=	4.14%,	$ADTFLAW_{1\text{-}15}$	=	9.68%
$IDTFLAW_{3\text{-}10}$	=	0.71%,	$IDTFLAW_{1\text{-}10}$	=	4.14%,	$IDTFLAW_{1\text{-}15}$	=	9.68%
$DFFLAW_{3\text{-}10}$	=	0.18%,	$DFFLAW_{1\text{-}10}$	=	1.04%,	$DFFLAW_{1\text{-}15}$	=	2.42%

The above results are documented in Table B-2.

Step 4B - Determine the Likelihood of Containment Breach Given Liner Flaw.

The likelihood of a breach in containment given a liner flaw is based on the Calvert Cliffs methodology [20] with a Pilgrim specific value for the upper-end pressure failure (100% likelihood) taken from Section 4.5 of the PSA [7]. A containment pressure of 113 psia corresponds with the 100% probability of failure. The lower-end pressure failure (0.1% likelihood) is set at 20 psia, consistent with Calvert Cliffs [20]. Per the Calvert Cliffs methodology [20], the containment failure probability (FP) versus containment pressure (P) is assumed to be an equation of the form:

$$FP(P) = b * e^{m*p}$$

Where:

FP (P) = containment failure probability given containment liner breach

m = slope of the containment failure probability

b = intercept of the containment failure probability

p = containment pressure, psia

The two anchor points of 0.1% at 20 psia and 100% at 113 psia provide sufficient information to solve for the slope m, and the intercept b, as follows:

Slope m,

$$m = LN(FP(100\%) - LN(0.1\%) / (\text{Upper Pressure} - \text{Lower Pressure})$$

$$m = LN(1.0) - LN(0.001) / (113\text{-}20)$$

$$m = 7.43 \times 10^{-2}$$

Intercept b,

$$b = FP(100\%) / e^{m*p}$$

$$b = 1 / e^{7.43 \times 10\text{-}2*113}$$

$$b = 2.25 \times 10^{-4}$$

The Pilgrim May 25, 1995 ILRT used a test pressure of 45.0 psig (or 59.7 psia) [21]. Based on this pressure the likelihood of containment breach in the liner is:

$$FP\ (59.7\ psia) = 2.25 \times 10^{-4} * e^{7.43 \times 10\text{-}2 * 59.7}$$

$$FP\ (59.7\ psia) = 0.0190 \quad or \quad 1.90\%$$

For the Drywell floor, the failure probability is set to one-tenth of the failure probability for Drywell walls, or 0.190%. (See Section B3.0, Assumption #4 and Section B4.0, Input #2).

Based on the above equation, containment liner breach and drywell floor intermediate values for FP are calculated and presented in Table B-3 and Figure B-1.

Step 5B - Determine the visual inspection detection failure.

This step examines the visual inspection detection failure likelihood for Pilgrim. The three areas of interest are the accessible portion of the drywell and torus, the inaccessible portion of the drywell and submergence area of the torus, and the inaccessible portion of the drywell floor.

The visual inspection detection failure likelihood for the accessible area of the drywell and torus (100% inside and outside of drywell head, 100% drywell liner inside, 100% torus outside area, and 100% torus inside area above waterline [22] is set to 10%, consistent with the Calvert Cliffs analysis [20]. This represents a 5% (0.05) failure to identify a visual flaw and 5% (0.05) likelihood that the flaw is not visible.

The inaccessible portion of the drywell (virtually 0% drywell liner outside because it is encased in concrete), and submergence area of the torus is assigned a 100% (1.0) visual detection failure likelihood. This is bounding, as the submerged area of the Torus may be examined.

Because the liner under the Drywell floor cannot be visually inspected, a visual detection failure likelihood of 100 % (1.0) is assigned, consistent with the Calvert Cliffs method.

The above results are documented in Table B-4.

Step 6B - Determine the likelihood of non-detected containment leakage

Per the Calvert Cliffs methodology [20], the likelihood of a non-detected containment leakage is calculated by multiplying the results of Steps 3B, 4B, and 5B. This yields the following:

Accessible portion of the drywell and torus,

$$ADTLEAK_{3\text{-}10} = ADTFLAW_{3\text{-}10} * ADTFP_{ILRT} * ADTVISUAL$$

$$ADTLEAK_{1\text{-}10} = ADTFLAW_{1\text{-}10} * ADTFP_{ILRT} * ADTVISUAL$$

$$ADTLEAK_{1\text{-}15} = ADTFLAW_{1\text{-}15} * ADTFP_{ILRT} * ADTVISUAL$$

Where:

$ADTLEAK_{3\text{-}10}$ = likelihood of non-detected containment leakage, given 3-in-10 years test interval and accessible portion of the drywell and torus

$ADTLEAK_{1\text{-}10}$ = likelihood of non-detected containment leakage, given 1-in-10 years test interval and accessible portion of the drywell and torus

$ADTLEAK_{1\text{-}15}$ = likelihood of non-detected containment leakage, given 1-in-15 years test interval and accessible portion of the drywell and torus

$ADTFLAW_{3\text{-}10}$ = increase in flaw likelihood at 3-in-10 years test interval given accessible portion of the drywell and torus = 0.71% (0.0071) [Table B-2]

$ADTFLAW_{1\text{-}10}$ = increase in flaw likelihood at 1-in-10 years test interval given accessible portion of the drywell and torus = 4.14% (0.0414) [Table B-2]

$ADTFLAW_{1\text{-}15}$ = increase in flaw likelihood at 1-in-15 years test interval given accessible portion of the drywell and torus = 9.68% (0.0968) [Table B-2]

$ADTFP_{ILRT}$ = likelihood of containment breach at ILRT test pressure (59.7 psia) given liner flaw and accessible portion of the drywell and torus = 0.0190 (1.90%) [Step 4B]

ADTVISUAL = visual inspection detection failure accessible portion of the drywell and torus
= 0.1 (10%) [Step 5B]

Therefore,

$$ADTLEAK_{3\text{-}10} = 0.0071 * 0.0190 * 0.1 = 1.349 \times 10^{-5} \ (0.001349\%)$$

$$ADTLEAK_{1\text{-}10} = 0.0414 * 0.0190 * 0.1 = 7.866 \times 10^{-5} \ (0.007866\%)$$

$$ADTLEAK_{1\text{-}15} = 0.0968 * 0.0190 * 0.1 = 1.839 \times 10^{-4} \ (0.018390\%)$$

Inaccessible portion of the drywell and submergence area of the torus,

$$IDTLEAK_{3\text{-}10} = IDTFLAW_{3\text{-}10} * ADTFP_{ILRT} * IDTVISUAL$$

$$IDTLEAK_{1\text{-}10} = IDTFLAW_{1\text{-}10} * ADTFP_{ILRT} * IDTVISUAL$$

$$IDTLEAK_{1\text{-}15} = IDTFLAW_{1\text{-}15} * ADTFP_{ILRT} * IDTVISUAL$$

Where:

$IDTLEAK_{3\text{-}10}$ = likelihood of non-detected containment leakage, given 3-in-10 years test interval and inaccessible portion of the drywell and submergence area of the torus

$IDTLEAK_{1\text{-}10}$ = likelihood of non-detected containment leakage, given 1-in-10 years test interval and inaccessible portion of the drywell and submergence area of the torus

$IDTLEAK_{1\text{-}15}$ = likelihood of non-detected containment leakage, given 1-in-15 years test interval and inaccessible portion of the drywell and submergence area of the torus

$IDTFLAW_{3\text{-}10}$ = increase in flaw likelihood at 3-in-10 years test interval given inaccessible portion of the drywell and submergence area of the torus = 0.71% (0.0071) [Table B-2]

$IDTFLAW_{1\text{-}10}$ = increase in flaw likelihood at 1-in-10 years test interval given inaccessible portion of the drywell and submergence area of the torus = 4.14% (0.0414) [Table B-2]

$IDTFLAW_{1\text{-}15}$ = increase in flaw likelihood at 1-in-15 years test interval given inaccessible portion of the drywell and submergence area of the torus = 9.68% (0.0968) [Table B-2]

$ADTFP_{ILRT}$ = likelihood of containment breach at ILRT test pressure (59.7 psia) given liner flaw and inaccessible portion of the drywell and submergence area of the torus
= 0.0190 (1.90%) [Step 4B]

IDTVISUAL = visual inspection detection failure inaccessible portion of the drywell and submergence area of the torus = 1.0 (100%) [Step 5B]

Therefore,

$IDTLEAK_{3\text{-}10}$ = 0.0071 * 0.0190 * 1.0 = 1.349×10^{-4} (0.01349%)

$IDTLEAK_{1\text{-}10}$ = 0.0414 * 0.0190 * 1.0 = 7.866×10^{-4} (0.07866%)

$IDTLEAK_{1\text{-}15}$ = 0.0968 * 0.0190 * 1.0 = 1.839×10^{-3} (0.18390%)

Inaccessible portion of the drywell floor,

$DFLEAK_{3\text{-}10}$ = $DFTFLAW_{3\text{-}10}$ * $DFTFP_{ILRT}$ * DFTVISUAL

$DFTLEAK_{1\text{-}10}$ = $DFTFLAW_{1\text{-}10}$ * $DFTFP_{ILRT}$ * DFTVISUAL

$DFTLEAK_{1\text{-}15}$ = $DFTFLAW_{1\text{-}15}$ * $DFTFP_{ILRT}$ * DFTVISUAL

Where:

$DFLEAK_{3\text{-}10}$ = likelihood of non-detected containment leakage, given 3-in-10 years test interval and inaccessible portion of the drywell floor

$DFLEAK_{1\text{-}10}$ = likelihood of non-detected containment leakage, given 1-in-10 years test interval and inaccessible portion of the drywell floor

$DFLEAK_{1\text{-}15}$ = likelihood of non-detected containment leakage, given 1-in-15 years test interval and inaccessible portion of the drywell floor

$DFFLAW_{3\text{-}10}$ = increase in flaw likelihood at 3-in-10 years test interval given inaccessible portion of the drywell floor = 0.18% (0.0018) [Table B-2]

$DFFLAW_{1\text{-}10}$ = increase in flaw likelihood at 1-in-10 years test interval given inaccessible portion of the drywell floor = 1.04% (0.0104) [Table B-2]

$DFFLAW_{1\text{-}15}$ = increase in flaw likelihood at 1-in-15 years test interval given inaccessible portion of the drywell floor = 2.42% (0.0242) [Table B-2]

$DFTFP_{ILRT}$ = likelihood of containment breach at ILRT test pressure (59.7 psia) given liner flaw and inaccessible portion of the drywell floor = 0.0019 (0.190%) [Step 4B]

DFVISUAL = visual inspection detection failure inaccessible portion of the drywell floor
= 1.0 (100%) [Step 5B]

Therefore,

$DFTLEAK_{3\text{-}10}$ = 0.0018 * 0.0019 * 1.0 = 3.420×10^{-6} (0.0003420%)
$DFTLEAK_{1\text{-}10}$ = 0.0104 * 0.0019 * 1.0 = 1.976×10^{-5} (0.001976%)
$DFTLEAK_{1\text{-}15}$ = 0.0242 * 0.0019 * 1.0 = 4.598×10^{-5} (0.004598%)

Total Likelihood of Non-Detected Containment Leakage due to Corrosion is,

$TOTAL_{3\text{-}10} = ADTLEAK_{3\text{-}10} + IDTLEAK_{3\text{-}10} + DFTLEAK_{3\text{-}10}$
$TOTAL_{1\text{-}10} = ADTLEAK_{1\text{-}10} + IDTLEAK_{1\text{-}10} + DFTLEAK_{1\text{-}10}$
$TOTAL_{1\text{-}15} = ADTLEAK_{1\text{-}15} + IDTLEAK_{1\text{-}15} + DFTLEAK_{1\text{-}15}$

Where:

$TOTAL_{3\text{-}10}$ = total likelihood of non-detected containment leakage due to corrosion, given 3-in-10 years test interval

$TOTAL_{1\text{-}10}$ = total likelihood of non-detected containment leakage due to corrosion, given 1-in-10 years test

$TOTAL_{1\text{-}15}$ = total likelihood of non-detected containment leakage due to corrosion, given 1-in-15 years test interval

$ADTLEAK_{3\text{-}10}$ = likelihood of non-detected containment leakage, given 3-in-10 years test interval and accessible portion of the drywell and torus

$ADTLEAK_{1\text{-}10}$ = likelihood of non-detected containment leakage, given 1-in-10 years test interval and accessible portion of the drywell and torus

$ADTLEAK_{1\text{-}15}$ = likelihood of non-detected containment leakage, given 1-in-15 years test interval and accessible portion of the drywell and torus

$IDTLEAK_{3\text{-}10}$ = likelihood of non-detected containment leakage, given 3-in-10 years test interval and inaccessible portion of the drywell and submergence area of the torus

$IDTLEAK_{1\text{-}10}$ = likelihood of non-detected containment leakage, given 1-in-10 years test interval and inaccessible portion of the drywell and submergence area of the torus

$IDTLEAK_{1\text{-}15}$ = likelihood of non-detected containment leakage, given 1-in-15 years test interval and inaccessible portion of the drywell and submergence area of the torus

$DFLEAK_{3\text{-}10}$ = likelihood of non-detected containment leakage, given 3-in-10 years test interval and inaccessible portion of the drywell floor

$DFLEAK_{1\text{-}10}$ = likelihood of non-detected containment leakage, given 1-in-10 years test interval and inaccessible portion of the drywell floor

$DFLEAK_{1\text{-}15}$ = likelihood of non-detected containment leakage, given 1-in-15 years test interval and inaccessible portion of the drywell floor

Therefore,

$$TOTAL_{3\text{-}10} = 0.001349\% + 0.01349\% + 0.0003420\% = 0.015181\%$$
$$TOTAL_{1\text{-}10} = 0.007866\% + 0.07866\% + 0.0019760\% = 0.088502\%$$
$$TOTAL_{1\text{-}15} = 0.018390\% + 0.18390\% + 0.0045980\% = 0.206888\%$$

The above results are documented in Table B-4.

Step 7B - Evaluate the Risk Impact in Terms of Population Dose Rate and Percentile Change for the Interval Cases.

This step calculates the change in population dose rate for EPRI accident Class 3b (all non-detectable containment failures are considered to result in large early releases), the change in percentage of the total dose rate attributable to liner corrosion and the change in this result dose rate from the base dose rate attributable to changes in ILRT surveillance interval.

The change in population dose rate is calculated as outline in Section 2.4.7 (Step 7), of this risk assessment (see page 43 of 77).

Increase to EPRI class 3b frequencies

$$LINER\_CLASS\_3b\_FREQ_{3\text{-}10} = (PROB_{class\_3b\_3\text{-}10} + LINER\_CLASS\_3B\_INCREASE_{3\text{-}10}) \times [CDF - (CDF_{LERF} + CDF_{NO\_LERF})]$$

$$LINER\_CLASS\_3b\_FREQ_{1\text{-}10} = (PROB_{class\_3b\_1\text{-}10} + LINER\_CLASS\_3B\_INCREASE_{1\text{-}10}) \times [CDF - (CDF_{LERF} + CDF_{NO\_LERF})]$$

$$LINER\_CLASS\_3b\_FREQ_{1\text{-}15} = (PROB_{class\_3b\_3\text{-}10} + LINER\_CLASS\_3B\_INCREASE_{1\text{-}15}) \times [CDF - (CDF_{LERF} + CDF_{NO\_LERF})]$$

Where:

$LINER\_CLASS\_3b\_FREQ_{3\text{-}10}$ = frequency of EPRI Class 3b due to liner corrosion failure given a 3-in-10 years ILRT interval

$LINER\_CLASS\_3b\_FREQ_{1\text{-}10}$ = frequency of EPRI Class 3b due to liner corrosion failure given a 1-in-10 years ILRT interval

$LINER\_CLASS\_3b\_FREQ_{1\text{-}15}$ = frequency of EPRI Class 3b due to liner corrosion failure given a 1-in-15 years ILRT interval

$PROB_{class\_3b\_3\text{-}10}$ = probability of large pre-existing containment liner leakage = 0.0027 [Section 2.3, input #9]

$PROB_{class\_3b\_1\text{-}10}$ = probability of large pre-existing containment liner leakage = 0.0090 [Section 2.4.5 Step 5, page 37 of 77]

$PROB_{class\_3b\_1\text{-}15}$ = probability of large pre-existing containment liner leakage = 0.0135 [Section 2.4.5 Step 5, page 38 of 77]

CDF_{LERF} = CDF for those individual sequences that independently cause a LERF $=1.13 \times 10^{-7}$/ry [Section 2.4.1, page 28 of 77]

CDF_{NO_LERF} = CDF for those individual sequences that never cause a LERF
= 5.86×10^{-6}/ry [Section 2.4.1, page 29 of 77]

CDF = Pilgrim Station PSA core damage frequency
= 6.41×10^{-6}/y [Section 2.3, input #2]

$LINER\_CLASS\_3B\_INCREASE_{3\text{-}10}$ = $TOTAL_{3\text{-}10}$ x EPRI_CLASS_3B_FRACTION

$LINER\_CLASS\_3B\_INCREASE_{1\text{-}10}$ = $TOTAL_{1\text{-}10}$ x EPRI_CLASS_3B_FRACTION

$LINER\_CLASS\_3B\_INCREASE_{1\text{-}15}$ = $TOTAL_{1\text{-}15}$ x EPRI_CLASS_3B_FRACTION

Where:
$LINER\_CLASS\_3B\_INCREASE_{3\text{-}10}$ = liner corrosion increase in EPRI class 3b given 3-in-10 years test interval

$LINER\_CLASS\_3B\_INCREASE_{1\text{-}10}$ = liner corrosion increase in EPRI class 3b given 1-in-10 years test interval

$LINER\_CLASS\_3B\_INCREASE_{1\text{-}15}$ = liner corrosion increase in EPRI class 3b given 1-in-15 years test interval

$TOTAL_{3\text{-}10}$ = total likelihood of non-detected containment leakage due to corrosion, given 3-in-10 years test interval
= 0.01518% [see above calculation and Table B-4]

$TOTAL_{1\text{-}10}$ = total likelihood of non-detected containment leakage due to corrosion, given 3-in-10 years test interval
= 0.08850% [see above calculation and Table B-4]

$TOTAL_{1\text{-}15}$ = total likelihood of non-detected containment leakage due to corrosion, given 3-in-10 years test interval
= 0.20689% [see above calculation and Table B-4]

EPRI_CLASS_3B_FRACTION = fraction of containment failures due to liner corrosion and considered to result in large early releases.
= 100% [Assumpiton#8]

Therefore:

$LINER\_CLASS\_3B\_INCREASE_{3\text{-}10}$ = 0.01518% x 1.0 = 0.01518%

$LINER\_CLASS\_3B\_INCREASE_{1\text{-}10}$ = 0.08850% x 1.0 = 0.08850%

$LINER\_CLASS\_3B\_INCREASE_{1\text{-}15}$ = 0.20689% x 1.0 = 0.20689%

Therefore:

$$LINER\_CLASS\_3b\_FREQ_{3\text{-}10} = (0.0027 + 0.01518\%) \times [6.41 \times 10^{-6}/ry - (1.13 \times 10^{-7}/ry + 5.86 \times 10^{-6}/ry)]$$

$$LINER\_CLASS\_3b\_FREQ_{3\text{-}10} = 1.24 \times 10^{-9}/ry$$

$$LINER\_CLASS\_3b\_FREQ_{1\text{-}10} = (0.0090 + 0.08850\%) \times [6.41 \times 10^{-6}/ry - (1.13 \times 10^{-7}/ry + 5.86 \times 10^{-6}/ry)]$$

$$LINER\_CLASS\_3b\_FREQ_{1\text{-}10} = 4.30 \times 10^{-9}/ry$$

$$LINER\_CLASS\_3b\_FREQ_{1\text{-}15} = (0.0135 + 0.20689\%) \times [6.41 \times 10^{-6}/ry - (1.13 \times 10^{-7}/ry + 5.86 \times 10^{-6}/ry)]$$

$$LINER\_CLASS\_3b\_FREQ_{1\text{-}15} = 6.80 \times 10^{-9}/ry$$

Increase to EPRI class 1 frequencies

$$LINER\_CLASS\_1\_FREQ_{3\text{-}10} = NCF - CLASS\_3a\_FREQUENCY - LINER\_CLASS\_3b\_FREQ_{3\text{-}10}$$

$$LINER\_CLASS\_1\_FREQ_{1\text{-}10} = NCF - CLASS\_3a\_FREQUENCY_{10} - LINER\_CLASS\_3b\_FREQ_{1\text{-}10}$$

$$LINER\_CLASS\_1\_FREQ_{1\text{-}15} = NCF - CLASS\_3a\_FREQUENCY_{15} - LINER\_CLASS\_3b\_FREQ_{1\text{-}15}$$

Where:

$LINER\_CLASS\_1\_FREQ_{3\text{-}10}$ = frequency of EPRI Class 1 given a 3-in-10 years ILRT interval

$LINER\_CLASS\_1\_FREQ_{1\text{-}10}$ = frequency of EPRI Class 1 given a 1-in-10 years ILRT interval

$LINER\_CLASS\_1\_FREQ_{1\text{-}15}$ = frequency of EPRI Class 1 given a 1-in-15 years ILRT interval

CLASS_3a_FREQUENCY = frequency of small pre-existing containment liner leakage
= $1.18 \times 10^{-8}/ry$ [Section 2.4.1 Step 1, page 29 of 77]

$CLASS\_3a\_FREQUENCY_{10}$ = frequency of small pre-existing containment liner leakage given a 1-in-10 years ILRT interval
= $3.93 \times 10^{-8}/ry$ [Section 2.4.5, Step 5 page 40 of 77]

$CLASS\_3a\_FREQUENCY_{15}$ = frequency of small pre-existing containment liner leakage given a 1-in-10 years ILRT interval
= $5.90 \times 10^{-8}/ry$ [Section 2.4.5, Step 5 page 40 of 77]

$LINER\_CLASS\_3b\_FREQ_{3\text{-}10}$ = frequency of EPRI Class 3b due to liner corrosion failure given a 3-in-10 years ILRT interval
= $1.24 \times 10^{-9}/ry$ [Above write-up, page B-12 of B-25]

$LINER\_CLASS\_3b\_FREQ_{1\text{-}10}$ = frequency of EPRI Class 3b due to liner corrosion failure given a 1-in-10 years ILRT interval
= $4.30 \times 10^{-9}/ry$ [Above write-up, page B-12 of B-25]

$LINER\_CLASS\_3b\_FREQ_{1\text{-}15}$ = frequency of EPRI Class 3b due to liner corrosion failure given a 3-in-15 years ILRT interval
= $6.80 \times 10^{-9}/ry$ [Above write-up, page B-12 of B-25]

NCF = frequency in which containment leakage is at or below maximum allowable Technical Specification Leakage
= $1.11 \times 10^{-7}/ry$ [Table 2-2]

Therefore:

$LINER\_CLASS\_1\_FREQ_{3\text{-}10} = 1.11 \times 10^{-7}/ry - 1.18 \times 10^{-8}/ry - 1.24 \times 10^{-9}/ry$
$LINER\_CLASS\_1\_FREQ_{3\text{-}10} = 9.80 \times 10^{-8}/ry$

$LINER\_CLASS\_1\_FREQ_{1\text{-}10} = 1.11 \times 10^{-7}/ry - 3.93 \times 10^{-8}/ry - 4.30 \times 10^{-9}/ry$
$LINER\_CLASS\_1\_FREQ_{1\text{-}10} = 6.76 \times 10^{-8}/ry$

$LINER\_CLASS\_1\_FREQ_{1\text{-}15} = 1.11 \times 10^{-7}/ry - 5.90 \times 10^{-8}/ry - 6.80 \times 10^{-9}/ry$
$LINER\_CLASS\_1\_FREQ_{1\text{-}15} = 4.55 \times 10^{-8}/ry$

The results of other pertinent calculations, are presented below as follows:

For 3-In-10 years,

EPRI Class	Person-rem	Frequency/Ry		Person-rem/Ry
1	1.06×10^{4}	9.80×10^{-8}		1.04×10^{-3}
2	4.53×10^{6}	4.42×10^{-11}	Corrosion Addition	2.00×10^{-4}
3a	1.06×10^{5}	1.18×10^{-8}		1.24×10^{-3}
3b	3.71×10^{5}	1.24×10^{-9}	6.61×10^{-11}	4.60×10^{-4}
4	N/A	0.0		0.0
5	N/A	0.0		0.0
6	N/A	0.0		0.0
7a	4.53×10^{6}	1.59×10^{-7}		7.19×10^{-1}
7b	1.82×10^{6}	2.19×10^{-8}		3.99×10^{-2}
7c	4.55×10^{6}	4.38×10^{-6}		1.99×10^{1}
7d	7.35×10^{5}	1.70×10^{-6}		1.25×10^{0}
8	5.66×10^{6}	3.79×10^{-8}		2.15×10^{-1}
Total		**6.41×10^{-6}**		**22.1568**

ILRT Dose Rate from 3a and 3b = 1.24×10^{-3} + 4.60 x 10-4 = 1.70×10^{-3} person-rem/ry
% Of Total = 100 * [1.24×10^{-3} + 4.60×10^{-4}] / 22.1568 = 0.0077%
LERF from 3b = 1.24×10^{-9} /ry
$CCFP\%_{LINER3\text{-}10}$ = 1 - [9.80×10^{-8} + 1.17×10^{-8}] / 6.41×10^{-6} = 98.29%

For 1-in-10 years.

EPRI Class	Person-rem	Frequency/Ry		Person-rem/Ry
1	1.06×10^{4}	6.76×10^{-8}		7.16×10^{-4}
2	4.53×10^{6}	4.42×10^{-11}	Corrosion Addition	2.00×10^{-4}
3a	1.06×10^{5}	3.93×10^{-8}		4.15×10^{-3}
3b	3.71×10^{5}	4.30×10^{-9}	3.85×10^{-10}	1.59×10^{-3}
4	N/A	0.0		0.0
5	N/A	0.0		0.0
6	N/A	0.0		0.0
7a	4.53×10^{6}	1.59×10^{-7}		7.19×10^{-1}
7b	1.82×10^{6}	2.19×10^{-8}		3.99×10^{-2}
7c	4.55×10^{6}	4.38×10^{-6}		1.99×10^{1}
7d	7.35×10^{5}	1.70×10^{-6}		1.25×10^{0}
8	5.66×10^{6}	3.79×10^{-8}		2.15×10^{-1}
Total		6.41×10^{-6}		22.1606

ILRT Dose Rate from 3a and 3b = 4.15×10^{-3} + 1.59×10^{-3} = 5.74×10^{-3} person-rem/ry
% Of Total = 100 * [4.15×10^{-3} + 1.59×10^{-3}] / 22.1606 = 0.0259%
LERF from 3b = 4.30×10^{-9} /ry
$CCFP\%_{LINER1-10}$ = 1 - [6.76×10^{-8} + 3.91×10^{-8}] / 6.41×10^{-6} = 98.34%

For 1-in-15 years.

EPRI Class	Person-rem	Frequency/Ry		Person-rem/Ry
1	1.06×10^{4}	4.55×10^{-8}		4.83×10^{-4}
2	4.53×10^{6}	4.42×10^{-11}	Corrosion Addition	2.00×10^{-4}
3a	1.06×10^{5}	5.90×10^{-8}		6.22×10^{-3}
3b	3.71×10^{5}	6.77×10^{-9}	8.99×10^{-10}	2.51×10^{-3}
4	N/A	0.0		0.0
5	N/A	0.0		0.0
6	N/A	0.0		0.0
7a	4.53×10^{6}	1.59×10^{-7}		7.19×10^{-1}
7b	1.82×10^{6}	2.19×10^{-8}		3.99×10^{-2}
7c	4.55×10^{6}	4.38×10^{-6}		1.99×10^{1}
7d	7.35×10^{5}	1.70×10^{-6}		1.25×10^{0}
8	5.66×10^{6}	3.79×10^{-8}		2.15×10^{-1}
Total		6.41×10^{-6}		22.1633

ILRT Dose Rate from 3a and 3b = 6.22×10^{-3} + 2.51×10^{-3} = 8.73×10^{-3} person-rem/ry
% Of Total = 100 * [6.22×10^{-3} + 2.51×10^{-3}] / 22.1633 = 0.0394%
LERF from 3b = 6.77×10^{-9} /ry
$CCFP\%_{LINER1-15}$ = 1 - [4.55×10^{-8} + 5.87×10^{-8}] / 6.41×10^{-6} = 98.37%

Based on the above results, the changes from the 1-in-10 years to 1-in-15 years dose rate is as follows:

$$INCREASE_{LINER10\text{-}15} = \left[\frac{TOT\text{-}DOSE_{RATE\text{-}LINER15} - TOT\text{-}DOSE_{RATE\text{-}LINER10}}{TOT\text{-}DOSE_{RATE\text{-}LINER10}} \right] * 100$$

Where:

$INCREASE_{LINER10\text{-}15}$ = percent change from 1-in-10 years ILRT interval to 1-in-15 years ILRT interval

$TOT\text{-}DOSE_{RATE\text{-}LINER15}$ = Total dose rate for all EPRI's Classes given a 1-in-15 years ILRT interval
= 22.1633 (person-rem/ry) [See for 1-in-15 years table above]

$TOT\text{-}DOSE_{RATE\text{-}LINER10}$ = Total dose rate for all EPRI's Classes given a 1-in-10 years ILRT interval
= 22.1606 (person-rem/ry) [See for 1-in-10 years table above]

Therefore,

$$INCREASE_{LINER10\text{-}15} = \left[\frac{22.1633 - 22.1606}{22.1606} \right] * 100 = 0.012\%$$

The above increase in risk on the total integrated plant risk for those accident sequences influenced by Type A testing, given the change from a 1-in-10 years test interval to a 1-in-15 years test interval, is found to be 0.012%. This value can be considered to be a negligible increase in risk.

Step 8B - Evaluate the risk impact in terms of LERF

This step calculates the change in the large early release frequency with extending the ILRT intervals from 1-in-10 years to 1-in15-years given the inclusion of a postulated liner corrosion flaw failure.

The affect on the LERF risk measure due to liner corrosion flaw is calculated as follows:

$$)LERF_{LNER10\text{-}15} = LINER\_CLASS\_3b\_FREQ_{1\text{-}15} - LINER\_CLASS\_3b\_FREQ_{1\text{-}10}$$

Where:

$)LERF_{LNER10\text{-}15}$ = the change in LERF from 1-in-10 years ILRT interval to 1-in-15 years ILRT interval

$LINER\_CLASS\_3b\_FREQ_{1\text{-}15}$ = frequency of EPRI accident Class 3b given a 1-in-15 years ILRT interval = 6.77×10^{-9}/ry [Step 7B]

$LINER\_CLASS\_3b\_FREQ_{1\text{-}10}$ = frequency of EPRI accident Class 3b given a 1-in-10 years ILRT interval = 4.30×10^{-9}/ry [Step 7B]

Therefore,

$$)LERF_{LNER10\text{-}15} = 6.77 \times 10^{-9} - 4.30 \times 10^{-9}$$

$$)LERF_{LNER10\text{-}15} = 2.47 \times 10^{-9}/ry$$

Based on this result, the inclusion of corrosion effects in the ILRT assessment would not change the previous conclusions of this calculation (See Sections 7 and 8). That is, the change in LERF from extending the interval to 15 years from the current 10 years requirement is estimated to be about 2.47×10^{-9}/ry. This value is below the NRC Regulatory Guide 1.174 [6] of 10^{-7}/yr. Therefore, because Regulatory Guide 1.174 [6] defines very small changes in LERF as below 10^{-7}/yr, increasing the ILRT interval at Pilgrim from the currently allowed 1-in-10 years to 1-in-15 years and taking into consideration the likelihood of a containment liner flaw due to corrosion is non-risk significant from a risk perspective.

Similarly, the change in LERF from the original 3-in-10-year interval is calculated as follows:

$$)LERF_{LNER3\text{-}15} = LINER\_CLASS\_3b\_FREQ_{1\text{-}15} - LINER\_CLASS\_3b\_FREQ_{3\text{-}10}$$

Where:

$)LERF_{LNER3\text{-}15}$ = the change in LERF from 3-in-10 years ILRT interval to 1-in-15 years ILRT interval

$LINER\_CLASS\_3b\_FREQ_{1\text{-}15}$ = frequency of EPRI accident Class 3b given a 1-in-15 years ILRT interval = 6.77×10^{-9}/ry [Step 7B]

$LINER\_CLASS\_3b\_FREQ_{3\text{-}10}$ = frequency of EPRI accident Class 3b given a 1-in-10 years ILRT interval = 1.24×10^{-9}/ry [Step 7B]

Therefore,

$$)LERF_{LNER3\text{-}15} = 6.77 \times 10^{-9} - 1.24 \times 10^{-9}$$

$$)LERF_{LNER3\text{-}15} = 5.53 \times 10^{-9}/ry$$

Similar to the $)LERF_{LNER10\text{-}15}$ result, the $)LERF_{LNER3\text{-}15}$ is also non-risk significant from a risk perspective.

Step 9B - Evaluate the change in conditional containment failure probability

This step calculates the change in conditional containment failure probability (CCFP). Similar to Section 2.4.9 Step 9 of this risk assessment, the change in CCFP tracts the impact of the ILRT on both early (LERF) and late radionuclide releases. Therefore, CCFP consists of all those accident sequences resulting in a radionuclide release other that the intact containment state for EPRI accident Class 1, and small failures state for EPRI accident Class 3a. In additional, the CCFP is conditional given a severe core damage accident. Therefore, the change in the conditional containment failure probability from 1-in-10 years to 1-in-15 years is:

$$)CCFP_{LINER10\text{-}15} = CCFP_{LINER1\text{-}15} - CCFP_{LINER1\text{-}10}$$

Where:

$)CCFP_{LINER10\text{-}15}$ = the change in conditional containment failure probability from 1-in-10 years to 1-in-15 years given non-detected containment leakage

$CCFP_{LINER1\text{-}10}$ = conditional containment failure probability given 1-in-10 years ILRT interval and potential non-detected containment leakage = 98.34% [Step 7B]

$CCFP_{LINER1\text{-}15}$ = conditional containment failure probability given 1-in-15 years ILRT interval and potential non-detected containment leakage = 98.37% [Step 7B]

Therefore,

$$)CCFP_{LINER10\text{-}15} = 98.37\% - 98.34\%$$

$$)CCFP_{LINER10\text{-}15} = 0.03\%$$

This change in $)CCFP_{LINER10\text{-}15}$ of less than 1% is insignificant from a risk perspective.

The results of Steps 7B, 8B, and 9B of the updated ILRT assessment including the potential impact from non-detected containment leakage scenarios assuming that 100% of the leakages result in EPRI Class 3b are show in Table B-5.

B6.0 Steel Shell Corrosion Sensitivity

Additional sensitivity cases were also developed to gain an understanding of the sensitivity of this analysis to the various key parameters. The sensitivity cases are as follows:

- Sensitivity Case 1 - Flaw rate doubles every 2 years
- Sensitivity Case 2 - Flaw rate doubles every 10 years
- Sensitivity Case 3 - 5% Visual inspection failures
- Sensitivity Case 4 - 15% Visual inspection failures
- Sensitivity Case 5 - Containment breach base point 10 times lower
- Sensitivity Case 6 - Containment breach base point 10 times higher
- Sensitivity Case 7 - Flaw rate doubles every 10 years, containment breach base point 10 times lower, 5% visual inspection failures and 10% EPRI accident Class 3b are LERF (Lower bound)
- Sensitivity Case 8 - Flaw rate doubles every 2 years, containment breach base point 10 times higher, 15% visual inspection failures and 100% EPRI accident Class 3b are LERF (upper bound)

The above sensitivities cases used the calculational methodology presented in Steps 2B to 9B. These steps were developed in an EXCEL spreadsheet. They are reproduced in Attachment A.

These results are summarized in Table B-6.

B7.0 Conclusions

This appendix provides a sensitivity evaluation of considering potential containment liner corrosion impacts within the structure of the ILRT interval extension risk assessment. The evaluation yields the following conclusions:

1. The impact of including age-adjusted corrosion effects in the ILRT assessment has minimal impact on plant risk and is therefore acceptable.

2. The change in LERF, taking into consideration the likelihood of a containment liner flaw due to age-adjusted corrosion is non-risk significant from a risk perspective. Specifically, extending the interval to 15 years from the current 10 years requirement is estimated to be about 2.47×10^{-9}/ry. This is below the Regulatory Guide 1.174 [6] acceptance criteria threshold of 10^{-7}/yr.

3. The age-adjusted corrosion impact in dose increase is estimated to be 2.70×10^{-3} person-rem/ry or 0.012% from the baseline ILRT 10 year's interval.

4. The age-adjusted corrosion impact on the conditional containment failure probability increase is estimated to be 0.3%.

5. A series of parametric sensitivity studies regarding potential age related corrosion effects on the containment steel liner also demonstrated minimal impact on plant risk.

Table B-1

Flaw Failure Rate as a Function of Time

Year	Accessible Area Drywell and Torus		Inaccessible Area Drywell and Torus		Drywell Floor	
	Failure Rate	Success Rate	Failure Rate	Success Rate	Failure Rate	Success Rate
0	1.79×10^{-3}	9.98×10^{-1}	1.79×10^{-3}	9.98×10^{-1}	4.46×10^{-4}	1.00
1	2.05×10^{-3}	9.98×10^{-1}	2.05×10^{-3}	9.98×10^{-1}	5.13×10^{-4}	9.99×10^{-1}
2	2.36×10^{-3}	9.98×10^{-1}	2.36×10^{-3}	9.98×10^{-1}	5.89×10^{-4}	9.99×10^{-1}
3	2.71×10^{-3}	9.97×10^{-1}	2.71×10^{-3}	9.97×10^{-1}	6.77×10^{-4}	9.99×10^{-1}
4	3.11×10^{-3}	9.97×10^{-1}	3.11×10^{-3}	9.97×10^{-1}	7.78×10^{-4}	9.99×10^{-1}
5	3.57×10^{-3}	9.96×10^{-1}	3.57×10^{-3}	9.96×10^{-1}	8.94×10^{-4}	9.99×10^{-1}
6	4.11×10^{-3}	9.96×10^{-1}	4.11×10^{-3}	9.96×10^{-1}	1.03×10^{-3}	9.99×10^{-1}
7	4.72×10^{-3}	9.95×10^{-1}	4.72×10^{-3}	9.95×10^{-1}	1.18×10^{-3}	9.99×10^{-1}
8	5.42×10^{-3}	9.95×10^{-1}	5.42×10^{-3}	9.95×10^{-1}	1.36×10^{-3}	9.99×10^{-1}
9	6.23×10^{-3}	9.94×10^{-1}	6.23×10^{-3}	9.94×10^{-1}	1.56×10^{-3}	9.98×10^{-1}
10	7.16×10^{-3}	9.93×10^{-1}	7.16×10^{-3}	9.93×10^{-1}	1.79×10^{-3}	9.98×10^{-1}
11	8.23×10^{-3}	9.92×10^{-1}	8.23×10^{-3}	9.92×10^{-1}	2.06×10^{-3}	9.98×10^{-1}
12	9.45×10^{-3}	9.91×10^{-1}	9.45×10^{-3}	9.91×10^{-1}	2.36×10^{-3}	9.98×10^{-1}
13	1.09×10^{-2}	9.89×10^{-1}	1.09×10^{-2}	9.89×10^{-1}	2.71×10^{-3}	9.97×10^{-1}
14	1.25×10^{-2}	9.88×10^{-1}	1.25×10^{-2}	9.88×10^{-1}	3.12×10^{-3}	9.97×10^{-1}
15	1.43×10^{-2}	9.86×10^{-1}	1.43×10^{-2}	9.86×10^{-1}	3.58×10^{-3}	9.96×10^{-1}

Table B-2

Flaw Failure Rate as a Function of Test Interval

Years	Accessible Area Drywell and Torus		Inaccessible Area Drywell and Torus		Drywell Floor	
	Failure Rate	Success Rate	Failure Rate	Success Rate	Failure Rate	Success Rate
3-in-10	0.71%	9.93×10^{-1}	0.71%	9.93×10^{-1}	0.18%	9.98×10^{-1}
1-in-10	4.14%	9.59×10^{-1}	4.14%	9.59×10^{-1}	1.04%	9.90×10^{-1}
1-in-15	9.68%	9.03×10^{-1}	9.68%	9.03×10^{-1}	2.42%	9.76×10^{-1}

Table B-3

Pilgrim Containment Failure Probability Given Containment Liner Flaw

Pressure (psia)	Containment Liner Failure Probability	Drywell Floor Failure Probability
0	0.0002	0.00002
10	0.0005	0.00005
15	0.0007	0.0001
20	0.0010	0.0001
30	0.0021	0.0002
40	0.0044	0.0004
50	0.0092	0.0009
60	0.0052	0.0005
70	0.0141	0.0014
80	0.0380	0.0038
90	0.1022	0.0102
95	0.1677	0.0168
100	0.2750	0.0275
105	0.4512	0.0451
110	0.7402	0.0740
111	0.8172	0.0817
112	0.9023	0.0902
113	0.9962	0.0996

Figure B-1 - Pilgrim Containment Failure Probability Given Containment Liner Flaw

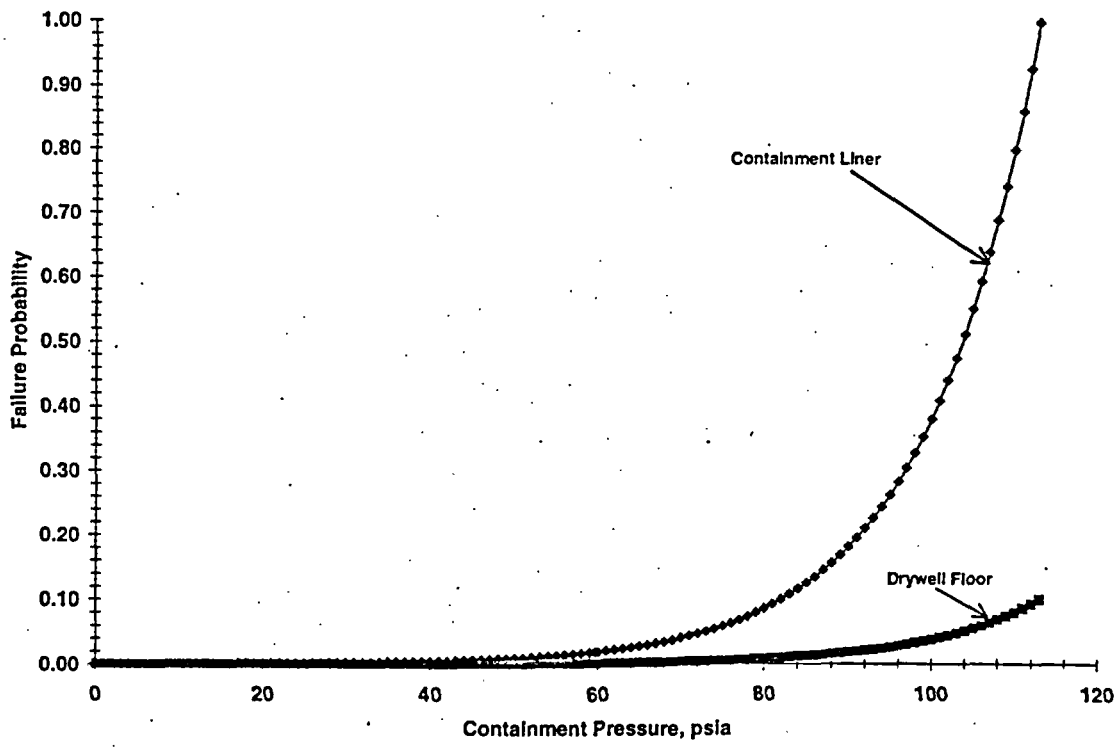


Table B-4

Pilgrim Containment Liner Corrosion Base Case

Step	Description	Accessible Area Drywell and Torus		Inaccessible Area Drywell and Torus		Drywell Floor	
1	Historical Steel Shell Flaw Likelihood	5.19×10^{-3}		5.19×10^{-3}		1.30×10^{-3}	
2	Age Adjusted Steel Shell Flaw Likelihood	Year	Failure Rate	Year	Failure Rate	Year	Failure Rate
		1	2.05×10^{-3}	1	2.05×10^{-3}	1	4.46×10^{-4}
		5-15	5.19×10^{-3}	5-15	5.19×10^{-3}	5-15	1.30×10^{-3}
		15	1.43×10^{-2}	15	1.43×10^{-2}	15	3.58×10^{-3}
3	Increase in Flaw Likelihood at 3, 10, and 15 years	0.71% (3-to-10 years) 4.14% (1-to-10 years) 9.68% (1-to-15 years)		0.71% (3-to-10 years) 4.14% (1-to-10 years) 9.68% (1-to-15 years)		0.18% (3-to-10 years) 1.04% (1-to-10 years) 2.42% (1-to-15 years)	
4	Likelihood of Breach in Containment Given Steel Shell Flaw	Pressure (psia)	Likelihood of Breach	Pressure (psia)	Likelihood of Breach	Pressure (psia)	Likelihood of Breach
		20	0.0010	20	0.0010	20	0.0001
		59.7 (ILRT)	0.0190	59.7 (ILRT)	0.0190	59.7 (ILRT)	0.0019
		100	0.3793	100	0.3793	100	0.0379
		110	0.7974	110	0.7974	120	0.0797
		113	0.9965	113	0.9965	155	0.0996
5	Visual Inspection Detection Failure Likelihood	0.1 (10%)		1.0 (100%)		1.0 (100%)	
6	Likelihood of Non-Detected Containment Leakage (Steps 3 * 4* 5)	0.00135% (3-to-10 years) 0.00787% (1-to-10 years) 0.01839% (1-to-15 years)		0.01349% (3-to-10 years) 0.07866% (1-to-10 years) 0.18390% (1-to-15 years)		0.00034% (3-to-10 years) 0.00198% (1-to-10 years) 0.00460% (1-to-15 years)	
Total Likelihood of Non-Detected Containment Leakage		0.01518% (3-to-10 years) 0.08850% (1-to-10 years) 0.20689% (1-to-15 years)					

Table B-5

Impact of Containment Steel Liner Corrosion on Pilgrim ILRT Intervals

	Base Case 3 Years			Extend to 10 Years			Extend to 15 Years		
EPRI Class	CDF (Per Ry)	Per-Rem	Per-Rem (Per Ry)	CDF (Per Ry)	Per-Rem	Per-Rem (Per Ry)	CDF (Per Ry)	Per-Rem	Per-Rem (Per Ry)
1	9.80×10^{-8}	1.06×10^{4}	1.04×10^{-3}	6.76×10^{-8}	1.06×10^{4}	7.16×10^{-4}	4.55×10^{-8}	1.06×10^{4}	4.83×10^{-4}
2	4.42×10^{-11}	4.53×10^{6}	2.00×10^{-4}	4.42×10^{-11}	4.53×10^{6}	2.00×10^{-4}	4.42×10^{-11}	4.53×10^{6}	2.00×10^{-4}
3a	1.18×10^{-8}	1.06×10^{5}	1.24×10^{-3}	3.93×10^{-8}	1.06×10^{5}	4.15×10^{-3}	5.90×10^{-8}	1.06×10^{5}	6.22×10^{-3}
3b	1.24×10^{-9}	3.71×10^{5}	4.60×10^{-4}	4.30×10^{-9}	3.71×10^{5}	1.59×10^{-3}	6.77×10^{-9}	3.71×10^{5}	2.51×10^{-3}
4	0.0	N/A	0.0	0.0	N/A	0.0	0.0	N/A	0.0
5	0.0	N/A	0.0	0.0	N/A	0.0	0.0	N/A	0.0
6	0.0	N/A	0.0	0.0	N/A	0.0	0.0	N/A	0.0
7b	1.59×10^{-7}	4.53×10^{6}	7.19×10^{-1}	1.59×10^{-7}	4.53×10^{6}	7.19×10^{-1}	1.59×10^{-7}	4.53×10^{6}	7.19×10^{-1}
7b	2.19×10^{-8}	1.82×10^{6}	3.99×10^{-2}	2.19×10^{-8}	1.82×10^{6}	3.99×10^{-2}	2.19×10^{-8}	1.82×10^{6}	3.99×10^{-2}
7c	4.38×10^{-6}	4.55×10^{6}	1.99×10^{1}	4.38×10^{-6}	4.55×10^{6}	1.99×10^{1}	4.38×10^{-6}	4.55×10^{6}	1.99×10^{1}
7d	1.70×10^{-6}	7.35×10^{5}	1.25×10^{0}	1.70×10^{-6}	7.35×10^{5}	1.25×10^{0}	1.70×10^{-6}	7.35×10^{5}	1.25×10^{0}
8	3.79×10^{-8}	5.66×10^{6}	2.15×10^{-1}	3.79×10^{-8}	5.66×10^{6}	2.15×10^{-1}	3.79×10^{-8}	5.66×10^{6}	2.15×10^{-1}
Total	6.41×10^{-6}		22.1568	6.41×10^{-6}		22.1606	6.41×10^{-6}		22.1633
ILRT Dose Rate from 3a and 3b			1.70×10^{-3} ($+2.45 \times 10^{-5}$)*			5.74×10^{-3} ($+1.43 \times 10^{-4}$)*			8.73×10^{-3} ($+3.34 \times 10^{-4}$)*
% Of Total			0.0077% (+0.0001%)*			0.0259% (+0.0006%)*			0.0394% (+0.0015%)*
Delta Dose Rate from 3a and 3b (10 to 15 yr)									2.70×10^{-3} (+0.0185%)*
LERF from 3b			1.24×10^{-9} ($+6.61 \times 10^{-11}$)*			4.30×10^{-9} ($+3.85 \times 10^{-10}$)*			6.77×10^{-9} ($+8.99 \times 10^{-10}$)*
Delta LERF (10 to 15 yr)									2.47×10^{-9} ($+5.14 \times 10^{-10}$)*
CCFP %			98.29% (+0.0010%)*			98.34% (+0.006%%)*			98.37% (+0.0140%)*
Delta CCFP % (10 to 15 yr)									0.03% (+0.0080%)*

* Denotes increase from original values presented in Steps 7, 8, and 9 of this calculation.

Table B-6

Containment Steel Liner Corrosion Sensitivity Cases

Age (Step 2)	Drywell/ Torus Breach (Step 4)	Visual Inspection & Non-Visual Flaws (Step 5)	Likelihood Flaw is LERF (EPRI Class 3b)	LERF Increase From Corrosion (3-in-10 years)	LERF Increase From Corrosion (1-in-10 years)	LERF Increase From Corrosion (1 to 15 years)	Total LERF Increase From ILRT Extension (10 to 15 years)
Base Case Doubles every 5 yrs	**Base Case** 1.8993%liner 0.1899%floor	**Base Case** 10%	**Base Case** 100%	**Base Case** 6.61×10^{-11}	**Base Case** 3.85×10^{-10}	**Base Case** 8.99×10^{-10}	**Base Case** **2.47×10^{-9}**
Doubles every 2 yrs	Base	Base	Base	1.89×10^{-11}	3.21×10^{-10}	1.86×10^{-9}	**3.50×10^{-9}**
Doubles every 10 yrs	Base	Base	Base	9.83×10^{-11}	1.35×10^{-10}	1.74×10^{-10}	**2.00×10^{-9}**
Base	Base	5%	Base	6.32×10^{-11}	3.68×10^{-10}	8.59×10^{-10}	**2.45×10^{-9}**
Base	Base	15%	Base	6.90×10^{-11}	4.02×10^{-10}	9.39×10^{-10}	**2.49×10^{-9}**
Base	0.5090%liner[12] 0.0509%floor[12]	Base	Base	1.77×10^{-11}	1.03×10^{-10}	2.41×10^{-10}	**2.09×10^{-9}**
Base	7.1249% liner[13] 0.7125%floor[13]	Base	Base	2.48×10^{-10}	1.44×10^{-9}	3.37×10^{-9}	**3.89×10^{-9}**
Lower Bound							
Doubles every 10 yrs	0.5090%liner[12] 0.0509%floor[12]	5%	10%	2.52×10^{-12}	1.09×10^{-11}	1.99×10^{-11}	**1.97×10^{-9}**
Upper Bound							
Doubles every 2 yrs	7.1249% liner[13] 0.7125%floor[13]	15%	100%	7.42×10^{-11}	1.26×10^{-9}	7.31×10^{-9}	**8.00×10^{-9}**

Attachment A

Pilgrim Risk Impact of Containment Liner Corrosion During an Extension of the ILRT Interval Results

Table of Contents

A1.0 Introduction

This attachment presents the results of the Pilgrim risk impact of containment liner corrosion during an extension of the ILRT interval. Eight sensitivity cases were examined. These are:

- Sensitivity Case 1 - Flaw rate doubles every 2 years
- Sensitivity Case 2 - Flaw rate doubles every 10 years
- Sensitivity Case 3 - 5% Visual inspection failures
- Sensitivity Case 4 - 15% Visual inspection failures
- Sensitivity Case 5 - Containment breach base point 10 times lower
- Sensitivity Case 6 - Containment breach base point 10 times higher
- Sensitivity Case 7 - Flaw rate doubles every 10 years, containment breach base point 10 times lower, 5% visual inspection failures and 10% EPRI accident Class 3b are LERF (Lower bound)
- Sensitivity Case 8 - Flaw rate doubles every 2 years, containment breach base point 10 times higher, 15% visual inspection failures and 100% EPRI accident Class 3b are LERF (upper bound)

The EXCEL spreadsheet results are presented in the following sections.

A2.0 Sensitivity Case 1 - Flaw Rate Doubles Every 2 Years

3-in-10 years

From Estimated Change

	Drywell/Torus	Inaccessible DW/Torus	Drywell Floor
1 to 3 years	0.20%	0.20%	0.05%
1 to 10 years	3.46%	3.46%	0.86%
1 to 15 years	20.07%	20.07%	5.02%
Other Assumptions:			
Containment Breach	1.8993%	1.8993%	0.1899%
Visual Inspection Failures	10.0%	100.0%	100.0%
EPRI Class 3a Fraction	0.0%	0.0%	0.0%
EPRI Class 3b Fraction	100.0%	100.0%	100.0%

Increases to 3a and 3b Frequencies	Drywell/Torus	Inaccessible DW/Torus	Drywell Floor	Total
	0.00000%	0.00000%	0.00000%	0.00000%
	0.00039%	0.00387%	0.00010%	0.00436%
				0.00436%

Release type	Pilgrim Dose Person-rem	CDF Frequency/ry	Case 3 in 10 yrs	Dose Person-rem/ry
1	1.06E+04	9.81E-08		1.04E-03
2	4.53E+06	4.42E-11	Corrosion Addition	2.00E-04
3a	1.06E+05	1.17E-08	0.00E+00	1.24E-03
3b	3.71E+05	1.19E-09	1.89E-11	4.43E-04
4	N/A	0.00E+00		0.00E+00
5	N/A	0.00E+00		0.00E+00
6	N/A	0.00E+00		0.00E+00
7a	4.53E+06	1.59E-07		7.19E-01
7b	1.82E+06	2.19E-08		3.99E-02
7c	4.55E+06	4.38E-06		1.99E+01
7d	7.35E+05	1.70E-06		1.25E+00
8	5.66E+06	3.79E-08		2.15E-01
Total		6.41E-06		22.1568

Risk Contribution:	0.01%
From 3a and 3b:	1.69E-03
3b LERF:	1.19E-09
CCFP:	98.29%

1-in-10 years

Increases to 3a and 3b Frequencies	Drywell/Torus	Inaccessible DW/Torus	Drywell Floor	Total
	0.0000%	0.0000%	0.0000%	0.00000%
	0.0066%	0.0656%	0.0016%	0.07384%
				0.07384%

Release type	Pilgrim Dose Person-rem	CDF Frequency/ry	Case 1 in 10 yrs	Dose Person-rem/ry
1	1.06E+04	6.76E-08		7.17E-04
2	4.53E+06	4.42E-11	Corrosion Addition	2.00E-04
3a	1.06E+05	3.91E-08	0.00E+00	4.15E-03
3b	3.71E+05	4.23E-09	3.21E-10	1.57E-03
4	N/A	0.00E+00		0.00E+00
5	N/A	0.00E+00		0.00E+00
6	N/A	0.00E+00		0.00E+00
7a	4.53E+06	1.59E-07		7.19E-01
7b	1.82E+06	2.19E-08		3.99E-02
7c	4.55E+06	4.38E-06		1.99E+01
7d	7.35E+05	1.70E-06		1.25E+00
8	5.66E+06	3.79E-08		2.15E-01
Total		6.41E-06		22.1605
			Risk Contribution:	0.03%
			From 3a and 3b:	5.72E-03
			3b LERF:	4.23E-09
			CCFP:	98.33%

1-in-15 years

Increases to 3a and 3b Frequencies	Drywell/Torus	Inaccessible DW/Torus	Drywell Floor	Total
	0.0000%	0.0000%	0.0000%	0.00000%
	0.0381%	0.3811%	0.0095%	0.42878%
				0.42878%

Release type	Pilgrim Dose Person-rem	CDF Frequency/ry	Case 1 in 15 yrs	Dose Person-rem/ry
1	1.06E+04	4.46E-08		4.72E-04
2	4.53E+06	4.42E-11	Corrosion Addition	2.00E-04
3a	1.06E+05	5.87E-08	0.00E+00	6.22E-03
3b	3.71E+05	7.73E-09	1.86E-09	2.87E-03
4	N/A	0.00E+00		0.00E+00
5	N/A	0.00E+00		0.00E+00
6	N/A	0.00E+00		0.00E+00
7a	4.53E+06	1.59E-07		7.19E-01
7b	1.82E+06	2.19E-08		3.99E-02
7c	4.55E+06	4.38E-06		1.99E+01
7d	7.35E+05	1.70E-06		1.25E+00
8	5.66E+06	3.79E-08		2.15E-01
Total		6.41E-06		22.1637
			Risk Contribution:	0.04%
			From 3a and 3b:	9.09E-03
			3b LERF:	7.73E-09
			CCFP:	98.39%

Other Pertinent Risk Metrics:

10 to 15 Increase (Person-rem/ry):	3.13E-03
3 to 15 Increase (Person-rem/ry):	6.84E-03
10 to 15 Delta-LERF:	3.50E-09
3 to 15 Delta-LERF:	6.54E-09
10 to 15 Delta-CCFP:	0.05%
3 to 15 Delta-CCFP:	0.10%
3 to 15 Delta-LERF from Corrosion:	1.85E-09
10 to 15 Delta-LERF from Corrosion:	1.54E-09
Increase in LERF (ILRT 3-to-15 years)	3.25E-08

A3.0 Sensitivity Case 2 - Flaw Rate Doubles Every 10 Years

3-in-10 years

From Estimated Change

	Drywell/Torus	Inaccessible DW/Torus	Drywell Floor
1 to 3 years	1.06%	1.06%	0.26%
1 to 10 years	4.58%	1.06%	1.15%
1 to 15 years	8.38%	1.06%	2.10%

Other Assumptions:

Containment Breach	1.8993%	1.8993%	0.1899%
Visual Inspection Failures	10.0%	100.0%	100.0%
EPRI Class 3a Fraction	0.0%	0.0%	0.0%
EPRI Class 3b Fraction	100.0%	100.0%	100.0%

Increases to 3a and 3b Frequencies

				Total
	0.00000%	0.00000%	0.00000%	0.00000%
	0.00201%	0.02010%	0.00050%	0.02261%
				0.02261%

Release type	Pilgrim Dose Person-rem	CDF Frequency/ry	Case 3 in 10 yrs	Dose Person-rem/ry
1	1.06E+04	9.80E-08		1.04E-03
2	4.53E+06	4.42E-11	Corrosion Addition	2.00E-04
3a	1.06E+05	1.17E-08	0.00E+00	1.24E-03
3b	3.71E+05	1.27E-09	9.83E-11	4.72E-04
4	N/A	0.00E+00		0.00E+00
5	N/A	0.00E+00		0.00E+00
6	N/A	0.00E+00		0.00E+00
7a	4.53E+06	1.59E-07		7.19E-01
7b	1.82E+06	2.19E-08		3.99E-02
7c	4.55E+06	4.38E-06		1.99E+01
7d	7.35E+05	1.70E-06		1.25E+00
8	5.66E+06	3.79E-08		2.15E-01
Total		6.41E-06		22.1569

Risk Contribution:	0.01%
From 3a and 3b:	1.72E-03
3b LERF:	1.27E-09
CCFP:	98.29%

1-in-10 years

Increases to 3a and 3b Frequencies	Drywell/Torus	Inaccessible DW/Torus	Drywell Floor	Total
	0.0000%	0.0000%	0.0000%	0.00000%
	0.0087%	0.0201%	0.0022%	0.03098%
				0.03098%

Release type	Pilgrim Dose Person-rem	CDF Frequency/ry	Case 1 in 10 yrs	Dose Person-rem/ry
1	1.06E+04	6.78E-08		7.19E-04
2	4.53E+06	4.42E-11	Corrosion Addition	2.00E-04
3a	1.06E+05	3.91E-08	0.00E+00	4.15E-03
3b	3.71E+05	4.05E-09	1.35E-10	1.50E-03
4	N/A	0.00E+00		0.00E+00
5	N/A	0.00E+00		0.00E+00
6	N/A	0.00E+00		0.00E+00
7a	4.53E+06	1.59E-07		7.19E-01
7b	1.82E+06	2.19E-08		3.99E-02
7c	4.55E+06	4.38E-06		1.99E+01
7d	7.35E+05	1.70E-06		1.25E+00
8	5.66E+06	3.79E-08		2.15E-01
Total		6.41E-06		22.1605
			Risk Contribution:	0.03%
			From 3a and 3b:	5.65E-03
			3b LERF:	4.05E-09
			CCFP:	98.33%

1-in-15 years

Increases to 3a and 3b Frequencies	Drywell/Torus	Inaccessible DW/Torus	Drywell Floor	Total
	0.0000%	0.0000%	0.0000%	0.00000%
	0.0159%	0.0201%	0.0040%	0.04001%
				0.04001%

Release type	Pilgrim Dose Person-rem	CDF Frequency/ry	Case 1 in 15 yrs	Dose Person-rem/ry
1	1.06E+04	4.63E-08		4.90E-04
2	4.53E+06	4.42E-11	Corrosion Addition	2.00E-04
3a	1.06E+05	5.87E-08	0.00E+00	6.22E-03
3b	3.71E+05	6.04E-09	1.74E-10	2.24E-03
4	N/A	0.00E+00		0.00E+00
5	N/A	0.00E+00		0.00E+00
6	N/A	0.00E+00		0.00E+00
7a	4.53E+06	1.59E-07		7.19E-01
7b	1.82E+06	2.19E-08		3.99E-02
7c	4.55E+06	4.38E-06		1.99E+01
7d	7.35E+05	1.70E-06		1.25E+00
8	5.66E+06	3.79E-08		2.15E-01
Total		6.41E-06		22.1631
			Risk Contribution:	0.04%
			From 3a and 3b:	8.47E-03
			3b LERF:	6.04E-09
			CCFP:	98.36%

Other Pertinent Risk Metrics:

10 to 15 Increase (Person-rem/ry):	2.59E-03
3 to 15 Increase (Person-rem/ry):	6.20E-03
10 to 15 Delta-LERF:	2.00E-09
3 to 15 Delta-LERF:	4.77E-09
10 to 15 Delta-CCFP:	0.03%
3 to 15 Delta-CCFP:	0.07%
3 to 15 Delta-LERF from Corrosion:	7.56E-11
10 to 15 Delta-LERF from Corrosion:	3.93E-11
Increase in LERF (ILRT 3-to-15 years)	6.00E-09

A4.0 Sensitivity Case 3 - 5% Visual Inspection Failures

3-in-10 years

From Estimated Change

	Drywell/Torus	Inaccessible DW/Torus	Drywell Floor
1 to 3 years	0.71%	0.71%	0.18%
1 to 10 years	4.14%	4.14%	1.04%
1 to 15 years	9.68%	9.68%	2.42%
Other Assumptions:			
Containment Breach	1.8993%	1.8993%	0.1899%
Visual Inspection Failures	5.0%	100.0%	100.0%
EPRI Class 3a Fraction	0.0%	0.0%	0.0%
EPRI Class 3b Fraction	100.0%	100.0%	100.0%

Increases to 3a and 3b Frequencies	Drywell/Torus	Inaccessible DW/Torus	Drywell Floor	Total
	0.0000%	0.0000%	0.0000%	0.00000%
	0.0007%	0.0135%	0.0003%	0.01453%
				0.01453%

Release type	Pilgrim Dose Person-rem	CDF Frequency/ry	Case 3 in 10 yrs	Dose Person-rem/ry
1	1.06E+04	9.80E-08		1.04E-03
2	4.53E+06	4.42E-11	Corrosion Addition	2.00E-04
3a	1.06E+05	1.17E-08	0.00E+00	1.24E-03
3b	3.71E+05	1.24E-09	6.32E-11	4.59E-04
4	N/A	0.00E+00		0.00E+00
5	N/A	0.00E+00		0.00E+00
6	N/A	0.00E+00		0.00E+00
7a	4.53E+06	1.59E-07		7.19E-01
7b	1.82E+06	2.19E-08		3.99E-02
7c	4.55E+06	4.38E-06		1.99E+01
7d	7.35E+05	1.70E-06		1.25E+00
8	5.66E+06	3.79E-08		2.15E-01
Total		6.41E-06		22.1568
			Risk Contribution:	0.01%
			From 3a and 3b:	1.70E-03
			3b LERF:	1.24E-09
			CCFP:	98.29%

1-in-10 years

Increases to 3a and 3b Frequencies	Drywell/Torus	Inaccessible DW/Torus	Drywell Floor	Total
	0.0000%	0.0000%	0.0000%	0.00000%
	0.0039%	0.0787%	0.0020%	0.08461%
				0.08461%

Release type	Pilgrim Dose Person-rem	CDF Frequency/ry	Case 1 in 10 yrs	Dose Person-rem/ry
1	1.06E+04	6.76E-08		7.16E-04
2	4.53E+06	4.42E-11	Corrosion Addition	2.00E-04
3a	1.06E+05	3.91E-08	0.00E+00	4.15E-03
3b	3.71E+05	4.28E-09	3.68E-10	1.59E-03
4	N/A	0.00E+00		0.00E+00
5	N/A	0.00E+00		0.00E+00
6	N/A	0.00E+00		0.00E+00
7a	4.53E+06	1.59E-07		7.19E-01
7b	1.82E+06	2.19E-08		3.99E-02
7c	4.55E+06	4.38E-06		1.99E+01
7d	7.35E+05	1.70E-06		1.25E+00
8	5.66E+06	3.79E-08		2.15E-01
Total		6.41E-06		22.1606
			Risk Contribution:	0.03%
			From 3a and 3b:	5.74E-03
			3b LERF:	4.28E-09
			CCFP:	98.34%

1-in-15 years

Increases to 3a and 3b Frequencies	Drywell/Torus	Inaccessible DW/Torus	Drywell Floor	Total
	0.0000%	0.0000%	0.0000%	0.00000%
	0.0092%	0.1838%	0.0046%	0.19761%
				0.19761%

Release type	Pilgrim Dose Person-rem	CDF Frequency/ry	Case 1 in 15 yrs	Dose Person-rem/ry
1	1.06E+04	4.56E-08		4.83E-04
2	4.53E+06	4.42E-11	Corrosion Addition	2.00E-04
3a	1.06E+05	5.87E-08	0.00E+00	6.22E-03
3b	3.71E+05	6.73E-09	8.59E-10	2.50E-03
4	N/A	0.00E+00		0.00E+00
5	N/A	0.00E+00		0.00E+00
6	N/A	0.00E+00		0.00E+00
7a	4.53E+06	1.59E-07		7.19E-01
7b	1.82E+06	2.19E-08		3.99E-02
7c	4.55E+06	4.38E-06		1.99E+01
7d	7.35E+05	1.70E-06		1.25E+00
8	5.66E+06	3.79E-08		2.15E-01
Total		6.41E-06		22.1633
			Risk Contribution:	0.04%
			From 3a and 3b:	8.72E-03
			3b LERF:	6.73E-09
			CCFP:	98.37%

Other Pertinent Risk Metrics:

10 to 15 Increase (Person-rem/ry):	2.75E-03
3 to 15 Increase (Person-rem/ry):	6.46E-03
10 to 15 Delta-LERF:	2.45E-09
3 to 15 Delta-LERF:	5.49E-09
10 to 15 Delta-CCFP:	0.04%
3 to 15 Delta-CCFP:	0.09%

3 to 15 Delta-LERF from Corrosion:	7.96E-10
10 to 15 Delta-LERF from Corrosion:	4.91E-10
Increase in LERF (ILRT 3-to-15 years)	1.90E-08

A5.0 Sensitivity Case 4 - 15% Visual Inspection Failures

3-in-10 years

From Estimated Change

	Drywell/Torus	Inaccessible DW/Torus	Drywell Floor
1 to 3 years	0.71%	0.71%	0.18%
1 to 10 years	4.14%	4.14%	1.04%
1 to 15 years	9.68%	9.68%	2.42%
Other Assumptions:			
Containment Breach	1.8993%	1.8993%	0.1899%
Visual Inspection Failures	15.0%	100.0%	100.0%
EPRI Class 3a Fraction	0.0%	0.0%	0.0%
EPRI Class 3b Fraction	100.0%	100.0%	100.0%

Increases to 3a and 3b Frequencies	Drywell/Torus	Inaccessible DW/Torus	Drywell Floor	Total
	0.0000%	0.0000%	0.0000%	0.00000%
	0.0020%	0.0135%	0.0003%	0.01588%
				0.01588%

Release type	Pilgrim Dose Person-rem	CDF Frequency/ry	Case 3 in 10 yrs	Dose Person-rem/ry
1	1.06E+04	9.80E-08		1.04E-03
2	4.53E+06	4.42E-11	Corrosion Addition	2.00E-04
3a	1.06E+05	1.17E-08	0.00E+00	1.24E-03
3b	3.71E+05	1.24E-09	6.90E-11	4.61E-04
4	N/A	0.00E+00		0.00E+00
5	N/A	0.00E+00		0.00E+00
6	N/A	0.00E+00		0.00E+00
7a	4.53E+06	1.59E-07		7.19E-01
7b	1.82E+06	2.19E-08		3.99E-02
7c	4.55E+06	4.38E-06		1.99E+01
7d	7.35E+05	1.70E-06		1.25E+00
8	5.66E+06	3.79E-08		2.15E-01
Total		6.41E-06		22.1568

Risk Contribution:	0.01%
From 3a and 3b:	1.71E-03
3b LERF:	1.24E-09
CCFP:	98.29%

1-in-10 years

Increases to 3a and 3b Frequencies	Drywell/Torus	Inaccessible DW/Torus	Drywell Floor	Total
	0.0000%	0.0000%	0.0000%	0.00000%
	0.0118%	0.0787%	0.0020%	0.09248%
				0.09248%

Release type	Pilgrim Dose Person-rem	CDF Frequency/ry	Case 1 in 10 yrs	Dose Person-rem/ry
1	1.06E+04	6.75E-08		7.16E-04
2	4.53E+06	4.42E-11	Corrosion Addition	2.00E-04
3a	1.06E+05	3.91E-08	0.00E+00	4.15E-03
3b	3.71E+05	4.32E-09	4.02E-10	1.60E-03
4	N/A	0.00E+00		0.00E+00
5	N/A	0.00E+00		0.00E+00
6	N/A	0.00E+00		0.00E+00
7a	4.53E+06	1.59E-07		7.19E-01
7b	1.82E+06	2.19E-08		3.99E-02
7c	4.55E+06	4.38E-06		1.99E+01
7d	7.35E+05	1.70E-06		1.25E+00
8	5.66E+06	3.79E-08		2.15E-01
Total		6.41E-06		22.1606
			Risk Contribution:	0.03%
			From 3a and 3b:	5.75E-03
			3b LERF:	4.32E-09
			CCFP:	98.34%

1-in-15 years

Increases to 3a and 3b Frequencies	Drywell/Torus	Inaccessible DW/Torus	Drywell Floor	Total
	0.0000%	0.0000%	0.0000%	0.00000%
	0.0276%	0.1838%	0.0046%	0.21600%
				0.21600%

Release type	Pilgrim Dose Person-rem	CDF Frequency/ry	Case 1 in 15 yrs	Dose Person-rem/ry
1	1.06E+04	4.55E-08		4.82E-04
2	4.53E+06	4.42E-11	Corrosion Addition	2.00E-04
3a	1.06E+05	5.87E-08	0.00E+00	6.22E-03
3b	3.71E+05	6.81E-09	9.39E-10	2.53E-03
4	N/A	0.00E+00		0.00E+00
5	N/A	0.00E+00		0.00E+00
6	N/A	0.00E+00		0.00E+00
7a	4.53E+06	1.59E-07		7.19E-01
7b	1.82E+06	2.19E-08		3.99E-02
7c	4.55E+06	4.38E-06		1.99E+01
7d	7.35E+05	1.70E-06		1.25E+00
8	5.66E+06	3.79E-08		2.15E-01
Total		6.41E-06		22.1633
			Risk Contribution:	0.04%
			From 3a and 3b:	8.75E-03
			3b LERF:	6.81E-09
			CCFP:	98.38%

Other Pertinent Risk Metrics:

10 to 15 Increase (Person-rem/ry):	2.77E-03
3 to 15 Increase (Person-rem/ry):	6.49E-03
10 to 15 Delta-LERF:	2.49E-09
3 to 15 Delta-LERF:	5.57E-09
10 to 15 Delta-CCFP:	0.04%
3 to 15 Delta-CCFP:	0.09%
3 to 15 Delta-LERF from Corrosion:	8.70E-10
10 to 15 Delta-LERF from Corrosion:	5.37E-10
Increase in LERF (ILRT 3-to-15 years)	2.08E-08

A6.0 Sensitivity Case 5 - Containment Breach Base Point 10 Times Lower

3-in-10 years

From Estimated Change

	Drywell/Torus	Inaccessible DW/Torus	Drywell Floor
1 to 3 years	0.71%	0.71%	0.18%
1 to 10 years	4.14%	4.14%	1.04%
1 to 15 years	9.68%	9.68%	2.42%
Other Assumptions:			
Containment Breach	0.5090%	0.5090%	0.0509%
Visual Inspection Failures	10.0%	100.0%	100.0%
EPRI Class 3a Fraction	0.0%	0.0%	0.0%
EPRI Class 3b Fraction	100.0%	100.0%	100.0%

Increases to 3a and 3b Frequencies	Drywell/Torus	Inaccessible DW/Torus	Drywell Floor	Total
	0.00000%	0.00000%	0.00000%	0.00000%
	0.00036%	0.00362%	0.00009%	0.00407%
				0.00407%

Release type	Pilgrim Dose Person-rem	CDF Frequency/ry	Case 3 in 10 yrs	Dose Person-rem/ry
1	1.06E+04	9.81E-08		1.04E-03
2	4.53E+06	4.42E-11	Corrosion Addition	2.00E-04
3a	1.06E+05	1.17E-08	0.00E+00	1.24E-03
3b	3.71E+05	1.19E-09	1.77E-11	4.42E-04
4	N/A	0.00E+00		0.00E+00
5	N/A	0.00E+00		0.00E+00
6	N/A	0.00E+00		0.00E+00
7a	4.53E+06	1.59E-07		7.19E-01
7b	1.82E+06	2.19E-08		3.99E-02
7c	4.55E+06	4.38E-06		1.99E+01
7d	7.35E+05	1.70E-06		1.25E+00
8	5.66E+06	3.79E-08		2.15E-01
Total		6.41E-06		22.1568

Risk Contribution:	0.0076%
From 3a and 3b:	1.69E-03
3b LERF:	1.19E-09
CCFP:	98.29%

1-in-10 years

Increases to 3a and 3b Frequencies	Drywell/Torus	Inaccessible DW/Torus	Drywell Floor	Total
	0.00000%	0.00000%	0.00000%	0.00000%
	0.00211%	0.02109%	0.00053%	0.02373%
				0.02373%

Release type	Pilgrim Dose Person-rem	CDF Frequency/ry	Case 1 in 10 yrs	Dose Person-rem/ry
1	1.06E+04	6.78E-08		7.19E-04
2	4.53E+06	4.42E-11	Corrosion Addition	2.00E-04
3a	1.06E+05	3.91E-08	0.00E+00	4.15E-03
3b	3.71E+05	4.02E-09	1.03E-10	1.49E-03
4	N/A	0.00E+00		0.00E+00
5	N/A	0.00E+00		0.00E+00
6	N/A	0.00E+00		0.00E+00
7a	4.53E+06	1.59E-07		7.19E-01
7b	1.82E+06	2.19E-08		3.99E-02
7c	4.55E+06	4.38E-06		1.99E+01
7d	7.35E+05	1.70E-06		1.25E+00
8	5.66E+06	3.79E-08		2.15E-01
Total		6.41E-06		22.1605
			Risk Contribution:	0.0254%
			From 3a and 3b:	5.64E-03
			3b LERF:	4.02E-09
			CCFP:	98.33%

1-in-15 years

Increases to 3a and 3b Frequencies	Drywell/Torus	Inaccessible DW/Torus	Drywell Floor	Total
	0.00000%	0.00000%	0.00000%	0.00000%
	0.00493%	0.04926%	0.00123%	0.05542%
				0.05542%

Release type	Pilgrim Dose Person-rem	CDF Frequency/ry	Case 1 in 15 yrs	Dose Person-rem/ry
1	1.06E+04	4.62E-08		4.90E-04
2	4.53E+06	4.42E-11	Corrosion Addition	2.00E-04
3a	1.06E+05	5.87E-08	0.00E+00	6.22E-03
3b	3.71E+05	6.11E-09	2.41E-10	2.27E-03
4	N/A	0.00E+00		0.00E+00
5	N/A	0.00E+00		0.00E+00
6	N/A	0.00E+00		0.00E+00
7a	4.53E+06	1.59E-07		7.19E-01
7b	1.82E+06	2.19E-08		3.99E-02
7c	4.55E+06	4.38E-06		1.99E+01
7d	7.35E+05	1.70E-06		1.25E+00
8	5.66E+06	3.79E-08		2.15E-01
Total		6.41E-06		22.1631
			Risk Contribution:	0.0383%
			From 3a and 3b:	8.49E-03
			3b LERF:	6.11E-09
			CCFP:	98.36%

Other Pertinent Risk Metrics:

10 to 15 Increase (Person-rem/ry):	2.62E-03
3 to 15 Increase (Person-rem/ry):	6.25E-03
10 to 15 Delta-LERF:	2.09E-09
3 to 15 Delta-LERF:	4.92E-09
10 to 15 Delta-CCFP:	0.033%
3 to 15 Delta-CCFP:	0.08%

3 to 15 Delta-LERF from Corrosion:	2.23E-10
10 to 15 Delta-LERF from Corrosion:	1.38E-10
Increase in LERF (ILRT 3-to-15 years)	5.33E-09

A7.0 Sensitivity Case 6 - Containment Breach Base Point 10 Times Higher

3-in-10 years

From Estimated Change

	Drywell/Torus	Inaccessible DW/Torus	Drywell Floor
1 to 3 years	0.71%	0.71%	0.18%
1 to 10 years	4.14%	4.14%	1.04%
1 to 15 years	9.68%	9.68%	2.42%
Other Assumptions:			
Containment Breach	7.1249%	7.1249%	0.7125%
Visual Inspection Failures	10.0%	100.0%	100.0%
EPRI Class 3a Fraction	0.0%	0.0%	0.0%
EPRI Class 3b Fraction	100.0%	100.0%	100.0%

Increases to 3a and 3b Frequencies	Drywell/Torus	Inaccessible DW/Torus	Drywell Floor	Total
	0.00000%	0.00000%	0.00000%	0.00000%
	0.00507%	0.05070%	0.00127%	0.05703%
				0.05703%

Release type	Pilgrim Dose Person-rem	CDF Frequency/ry	Case 3 in 10 yrs	Dose Person-rem/ry
1	1.06E+04	4.42E-11	Corrosion Addition	2.00E-04
2	4.53E+06	1.17E-08	0.00E+00	1.24E-03
3a	1.06E+05	1.42E-09	2.48E-10	5.28E-04
3b	3.71E+05	0.00E+00		0.00E+00
4	N/A	0.00E+00		0.00E+00
5	N/A	0.00E+00		0.00E+00
6	N/A	1.59E-07		7.19E-01
7a	4.53E+06	2.19E-08		3.99E-02
7b	1.82E+06	4.38E-06		1.99E+01
7c	4.55E+06	1.70E-06		1.25E+00
7d	7.35E+05	3.79E-08		2.15E-01
8	5.66E+06	6.41E-06		22.1569
Total		4.42E-11	Corrosion Addition	2.00E-04
			Risk Contribution:	0.0080%
			From 3a and 3b:	1.77E-03
			3b LERF:	1.42E-09
			CCFP:	98.29%

1-in-10 years

Increases to 3a and 3b Frequencies	Drywell/Torus	Inaccessible DW/Torus	Drywell Floor	Total
	0.00000%	0.00000%	0.00000%	0.00000%
	0.02953%	0.29525%	0.00738%	0.33216%
				0.33216%

Release type	Pilgrim Dose Person-rem	CDF Frequency/ry	Case 1 in 10 yrs	Dose Person-rem/ry
1	1.06E+04	6.65E-08		7.05E-04
2	4.53E+06	4.42E-11	Corrosion Addition	2.00E-04
3a	1.06E+05	3.91E-08	0.00E+00	4.15E-03
3b	3.71E+05	5.36E-09	1.44E-09	1.99E-03
4	N/A	0.00E+00		0.00E+00
5	N/A	0.00E+00		0.00E+00
6	N/A	0.00E+00		0.00E+00
7a	4.53E+06	1.59E-07		7.19E-01
7b	1.82E+06	2.19E-08		3.99E-02
7c	4.55E+06	4.38E-06		1.99E+01
7d	7.35E+05	1.70E-06		1.25E+00
8	5.66E+06	3.79E-08		2.15E-01
Total		6.41E-06		22.1609
			Risk Contribution:	0.0277%
			From 3a and 3b:	6.14E-03
			3b LERF:	5.36E-09
			CCFP:	98.35%

1-in-15 years

Increases to 3a and 3b Frequencies	Drywell/Torus	Inaccessible DW/Torus	Drywell Floor	Total
	0.00000%	0.00000%	0.00000%	0.00000%
	0.06896%	0.68961%	0.01724%	0.77581%
				0.77581%

Release type	Pilgrim Dose Person-rem	CDF Frequency/ry	Case 1 in 15 yrs	Dose Person-rem/ry
1	1.06E+04	4.31E-08		4.56E-04
2	4.53E+06	4.42E-11	Corrosion Addition	2.00E-04
3a	1.06E+05	5.87E-08	0.00E+00	6.22E-03
3b	3.71E+05	9.24E-09	3.37E-09	3.43E-03
4	N/A	0.00E+00		0.00E+00
5	N/A	0.00E+00		0.00E+00
6	N/A	0.00E+00		0.00E+00
7a	4.53E+06	1.59E-07		7.19E-01
7b	1.82E+06	2.19E-08		3.99E-02
7c	4.55E+06	4.38E-06		1.99E+01
7d	7.35E+05	1.70E-06		1.25E+00
8	5.66E+06	3.79E-08		2.15E-01
Total		6.41E-06		22.1642
			Risk Contribution:	0.0435%
			From 3a and 3b:	9.65E-03
			3b LERF:	9.24E-09
			CCFP:	98.41%

Other Pertinent Risk Metrics:

10 to 15 Increase (Person-rem/ry):	3.27E-03
3 to 15 Increase (Person-rem/ry):	7.30E-03
10 to 15 Delta-LERF:	3.89E-09
3 to 15 Delta-LERF:	7.82E-09
10 to 15 Delta-CCFP:	0.061%
3 to 15 Delta-CCFP:	0.12%
3 to 15 Delta-LERF from Corrosion:	3.13E-09
10 to 15 Delta-LERF from Corrosion:	1.93E-09
Increase in LERF (ILRT 3-to-15 years)	7.47E-08

A8.0 Sensitivity Case 7 - Lower bound
(Flaw rate doubles every 10 years, containment breach base point 10 times lower, 5% visual inspection failures and 10% EPRI accident Class 3b are LERF)

3-in-10 years

From Estimated Change

	Drywell/Torus	Inaccessible DW/Torus	Drywell Floor
1 to 3 years	1.06%	1.06%	0.26%
1 to 10 years	4.58%	4.58%	1.15%
1 to 15 years	8.38%	8.38%	2.10%
Other Assumptions:			
Containment Breach	0.5090%	0.5090%	0.0509%
Visual Inspection Failures	5.0%	100.0%	100.0%
EPRI Class 3a Fraction	90.0%	90.0%	90.0%
EPRI Class 3b Fraction	10.0%	10.0%	10.0%

Increases to 3a and 3b Frequencies	Drywell/Torus	Inaccessible DW/Torus	Drywell Floor	Total
	0.00024%	0.00485%	0.00012%	0.00521%
	0.00003%	0.00054%	0.00001%	0.00058%
				0.00579%

Release type	Pilgrim Dose Person-rem	CDF Frequency/ry	Case 3 in 10 yrs	Dose Person-rem/ry
1	1.06E+04	9.81E-08		1.04E-03
2	4.53E+06	4.42E-11	Corrosion Addition	2.00E-04
3a	1.06E+05	1.18E-08	2.27E-11	1.25E-03
3b	3.71E+05	1.18E-09	2.52E-12	4.37E-04
4	N/A	0.00E+00		0.00E+00
5	N/A	0.00E+00		0.00E+00
6	N/A	0.00E+00		0.00E+00
7a	4.53E+06	1.59E-07		7.19E-01
7b	1.82E+06	2.19E-08		3.99E-02
7c	4.55E+06	4.38E-06		1.99E+01
7d	7.35E+05	1.70E-06		1.25E+00
8	5.66E+06	3.79E-08		2.15E-01
Total		6.41E-06		22.1568
			Risk Contribution:	0.0076%
			From 3a and 3b:	1.68E-03
			3b LERF:	1.18E-09
			CCFP:	98.29%

1-in-10 years

Increases to 3a and 3b Frequencies	Drywell/Torus	Inaccessible DW/Torus	Drywell Floor	Total
	0.00105%	0.02099%	0.00052%	0.02257%
	0.00012%	0.00233%	0.00006%	0.00251%
				0.02507%

Release type	Pilgrim Dose Person-rem	CDF Frequency/ry	Case 1 in 10 yrs	Dose Person-rem/ry
1	1.06E+04	6.78E-08		7.19E-04
2	4.53E+06	4.42E-11	Corrosion Addition	2.00E-04
3a	1.06E+05	3.92E-08	9.81E-11	4.16E-03
3b	3.71E+05	3.92E-09	1.09E-11	1.46E-03
4	N/A	0.00E+00		0.00E+00
5	N/A	0.00E+00		0.00E+00
6	N/A	0.00E+00		0.00E+00
7a	4.53E+06	1.59E-07		7.19E-01
7b	1.82E+06	2.19E-08		3.99E-02
7c	4.55E+06	4.38E-06		1.99E+01
7d	7.35E+05	1.70E-06		1.25E+00
8	5.66E+06	3.79E-08		2.15E-01
Total		6.41E-06		22.1604
			Risk Contribution:	0.0253%
			From 3a and 3b:	5.61E-03
			3b LERF:	3.92E-09
			CCFP:	98.33%

1-in-15 years

Increases to 3a and 3b Frequencies	Drywell/Torus	Inaccessible DW/Torus	Drywell Floor	Total
	0.00192%	0.03841%	0.00096%	0.04129%
	0.00021%	0.00427%	0.00011%	0.00459%
				0.04588%

Release type	Pilgrim Dose Person-rem	CDF Frequency/ry	Case 1 in 15 yrs	Dose Person-rem/ry
1	1.06E+04	4.62E-08		4.90E-04
2	4.53E+06	4.42E-11	Corrosion Addition	2.00E-04
3a	1.06E+05	5.89E-08	1.80E-10	6.24E-03
3b	3.71E+05	5.89E-09	1.99E-11	2.19E-03
4	N/A	0.00E+00		0.00E+00
5	N/A	0.00E+00		0.00E+00
6	N/A	0.00E+00		0.00E+00
7a	4.53E+06	1.59E-07		7.19E-01
7b	1.82E+06	2.19E-08		3.99E-02
7c	4.55E+06	4.38E-06		1.99E+01
7d	7.35E+05	1.70E-06		1.25E+00
8	5.66E+06	3.79E-08		2.15E-01
Total		6.41E-06		22.1630
			Risk Contribution:	0.0380%
			From 3a and 3b:	8.43E-03
			3b LERF:	5.89E-09
			CCFP:	98.36%

Other Pertinent Risk Metrics:

10 to 15 Increase (Person-rem/ry):	2.58E-03
3 to 15 Increase (Person-rem/ry):	6.19E-03
10 to 15 Delta-LERF:	1.97E-09
3 to 15 Delta-LERF:	4.71E-09
10 to 15 Delta-CCFP:	0.031%
3 to 15 Delta-CCFP:	0.07%
3 to 15 Delta-LERF from Corrosion:	1.74E-11
10 to 15 Delta-LERF from Corrosion:	9.05E-12
Increase in LERF (ILRT 3-to-15 years)	4.92E-09

A9.0 Sensitivity Case 8 - Upper Bound
(Flaw rate doubles every 2 years, containment breach base point 10 times higher, 15% visual inspection failures and 100% EPRI accident Class 3b are LERF)

3-in-10 years

From Estimated Change

	Drywell/Torus	Inaccessible DW/Torus	Drywell Floor
1 to 3 years	0.20%	0.20%	0.05%
1 to 10 years	3.46%	3.46%	0.86%
1 to 15 years	20.07%	20.07%	5.02%
Other Assumptions:			
Containment Breach	7.1249%	7.1249%	0.7125%
Visual Inspection Failures	15.0%	100.0%	100.0%
EPRI Class 3a Fraction	0.0%	0.0%	0.0%
EPRI Class 3b Fraction	100.0%	100.0%	100.0%

Increases to 3a and 3b Frequencies	Drywell/Torus	Inaccessible DW/Torus	Drywell Floor	Total
	0.00000%	0.00000%	0.00000%	0.00000%
	0.00218%	0.01452%	0.00036%	0.01706%
				0.01706%

Release type	Pilgrim Dose Person-rem	CDF Frequency/ry	Case 3 in 10 yrs	Dose Person-rem/ry
1	1.06E+04	9.80E-08		1.04E-03
2	4.53E+06	4.42E-11	Corrosion Addition	2.00E-04
3a	1.06E+05	1.17E-08	0.00E+00	1.24E-03
3b	3.71E+05	1.25E-09	7.42E-11	4.63E-04
4	N/A	0.00E+00		0.00E+00
5	N/A	0.00E+00		0.00E+00
6	N/A	0.00E+00		0.00E+00
7a	4.53E+06	1.59E-07		7.19E-01
7b	1.82E+06	2.19E-08		3.99E-02
7c	4.55E+06	4.38E-06		1.99E+01
7d	7.35E+05	1.70E-06		1.25E+00
8	5.66E+06	3.79E-08		2.15E-01
Total		6.41E-06		22.1568
			Risk Contribution:	0.0077%
			From 3a and 3b:	1.71E-03
			3b LERF:	1.25E-09
			CCFP:	98.29%

1-in-10 years

Increases to 3a and 3b Frequencies	Drywell/Torus	Inaccessible DW/Torus	Drywell Floor	Total
	0.00000%	0.00000%	0.00000%	0.00000%
	0.03693%	0.24622%	0.00616%	0.28930%
				0.28930%

Release type	Pilgrim Dose Person-rem	CDF Frequency/ry	Case 1 in 10 yrs	Dose Person-rem/ry
1	1.06E+04	6.67E-08		7.07E-04
2	4.53E+06	4.42E-11	Corrosion Addition	2.00E-04
3a	1.06E+05	3.91E-08	0.00E+00	4.15E-03
3b	3.71E+05	5.17E-09	1.26E-09	1.92E-03
4	N/A	0.00E+00		0.00E+00
5	N/A	0.00E+00		0.00E+00
6	N/A	0.00E+00		0.00E+00
7a	4.53E+06	1.59E-07		7.19E-01
7b	1.82E+06	2.19E-08		3.99E-02
7c	4.55E+06	4.38E-06		1.99E+01
7d	7.35E+05	1.70E-06		1.25E+00
8	5.66E+06	3.79E-08		2.15E-01
Total		6.41E-06		22.1609
			Risk Contribution:	0.0274%
			From 3a and 3b:	6.07E-03
			3b LERF:	5.17E-09
			CCFP:	98.35%

1-in-15 years

Increases to 3a and 3b Frequencies	Drywell/Torus	Inaccessible DW/Torus	Drywell Floor	Total
	0.00000%	0.00000%	0.00000%	0.00000%
	0.21447%	1.42979%	0.03574%	1.68000%
				1.68000%

Release type	Pilgrim Dose Person-rem	CDF Frequency/ry	Case 1 in 15 yrs	Dose Person-rem/ry
1	1.06E+04	3.91E-08		4.15E-04
2	4.53E+06	4.42E-11	Corrosion Addition	2.00E-04
3a	1.06E+05	5.87E-08	0.00E+00	6.22E-03
3b	3.71E+05	1.32E-08	7.31E-09	4.89E-03
4	N/A	0.00E+00		0.00E+00
5	N/A	0.00E+00		0.00E+00
6	N/A	0.00E+00		0.00E+00
7a	4.53E+06	1.59E-07		7.19E-01
7b	1.82E+06	2.19E-08		3.99E-02
7c	4.55E+06	4.38E-06		1.99E+01
7d	7.35E+05	1.70E-06		1.25E+00
8	5.66E+06	3.79E-08		2.15E-01
Total		6.41E-06		22.1656
			Risk Contribution:	0.0501%
			From 3a and 3b:	1.11E-02
			3b LERF:	1.32E-08
			CCFP:	98.47%

Other Pertinent Risk Metrics:

10 to 15 Increase (Person-rem/ry):	4.75E-03
3 to 15 Increase (Person-rem/ry):	8.78E-03
10 to 15 Delta-LERF:	8.00E-09
3 to 15 Delta-LERF:	1.19E-08
10 to 15 Delta-CCFP:	0.125%
3 to 15 Delta-CCFP:	0.19%
3 to 15 Delta-LERF from Corrosion:	7.23E-09
10 to 15 Delta-LERF from Corrosion:	6.05E-09
Increase in LERF (ILRT 3-to-15 years)	1.27E-07

Attachment 5 to 2.04.027

Entergy Nuclear Operations, Inc.
Pilgrim Nuclear Power Plant
Proposed Amendment to the Technical Specifications

Pilgrim Nuclear Power Station Procedure QA 20.03, Rev.2
"First Ten-Year Interval IWE Containment Inspection Program"
(29 pages)

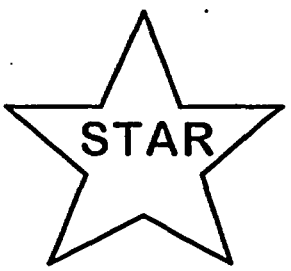
Attachment 5 to 2.04.027

PILGRIM NUCLEAR POWER STATION

Procedure No. QA20.03

FIRST TEN-YEAR INTERVAL

IWE CONTAINMENT INSPECTION PROGRAM



Stop
Think
Act
Review

ENGINEERING RELATED

QA PROGRAM RELATED

QAI RELATED

REVISION LOG

REVISION 2 **Date Originated 9/02**

Pages Affected	Description
4	Add QC Inspection Report to References.
7	Revise text of Drywell Shell, Code Item E4.12 in Discussion.
11	Delete Drywell Shell at Sand Cushion Region from Table 5.3.1. Revise number of Components/Areas and Inspection Notes for Item Number E4.12.
12	Revise Item number E5.30 number of Components/Areas of Table 5.3.1.
14,15	Revise Table 5.3.2 IWE period for various components.

REVISION 1 **Date Originated 12/99**

Pages Affected	Description
5	Include reference to NRC approval of Relief Requests.
12,15	Correct statements regarding moisture barrier at 9' and delete same barrier from Table 5.3.2.

REVISION 0 **Date Originated 1/99**

Pages Affected	Description
All	New Procedure required to document the IWE Containment Inspection Program for PNPS.

TABLE OF CONTENTS

1.0 PURPOSE AND SCOPE

This Procedure defines the First Ten-Year Interval IWE Containment Inspection Program.

2.0 REFERENCES

[1] 10CFR50.55a(b)(2)

[2] ASME Boiler and Pressure Vessel Code, Section XI, 1992 Edition with 1992 Addenda

[3] QC Inspection Report IR 02-0400, Deletion of Drywell UT Exams from IWE Program

3.0 DEFINITIONS

None

4.0 RESPONSIBILITIES

None

5.0 DISCUSSION

5.1 INTRODUCTION AND PLAN DESCRIPTION

5.1.1 Overview

[1] This Containment Inservice Inspection Plan outlines the requirements for the inspection of Class MC pressure retaining components (Primary Containment) and their integral attachments at the Pilgrim Nuclear Power Station (PNPS). The Plan details inservice inspection requirements for Class MC components in accordance with the requirements of 10CFR50.55a(b)(2) and the 1992 edition of ASME Boiler and Pressure Vessel Code Section XI with 1992 addenda, Inspection Program B.

[2] This Inservice Inspection Plan is effective from September 9, 1998, through and including September 9, 2008, with the first examinations taking place in 1999. This time period represents the First Ten-Year Interval for IWE containment inspections at PNPS.

[3] Containment inservice examinations scheduled for the first 40-month period of the ten-year IWE inspection interval shall be completed by September 9, 2001, as required by the regulation for expedited examinations. These examinations shall serve the same purpose as preservice baseline examinations.

[4] Submittal of this Containment Inspection Plan to the Nuclear Regulatory Commission for approval is not required, but shall remain available on site for audit purposes as required.

[5] The main features of this Plan are the Introduction and Overview, Relief Requests, Summary Tables, and Scheduled Examination Tables. Additional information such as the Program Drawing index and NRC Correspondence index are also included.

5.1.2 Basis of Inservice Inspection Plan

[1] The Plan is based on the requirements of 10CFR50.55a(b)(2) and the 1992 edition of ASME Section XI with 1992 addenda, subsections IWA and IWE only. Relief has been requested and granted from those portions of the inspection Code that would constitute a burden to PNPS without a compensating increase in quality and safety or are considered impractical. Relief Requests are included in Section 5.4 of this Procedure. The Design/Fabrication Code for the Pilgrim Station BWR Mark I containment is ASME Section III 1965 edition and the latest addenda as of June 9, 1967, including Code Cases 1330-1 and 1177-5. The containment vessel is a Class "B" vessel as defined in the above code.

[2] Although not required by the regulation, containment supports shall be examined in accordance with the 1989 edition of ASME Section XI as modified by Code Case N-491.

[3] The optional Category E-B examinations for Pressure Retaining Welds and Category E-F examinations for Pressure Retaining Dissimilar Metal Welds are not included in this Inspection Plan.

[4] For inservice examinations of Class MC components that reveal flaws or areas of degradation exceeding the acceptance standards of Table IWE-3410-1, the provisions of 10CFR50.55a(b)(2)(x)(D) shall be used as an alternative to the additional examination provisions (scope expansion) of ASME XI subparagraph IWE-2430.

[5] The General Visual Examination shall be scheduled during each 40-month inspection period to coincide with the dates of the Appendix J containment walkdowns typically performed prior to each Appendix J Type A test.

[6] The designated Responsible Engineer required by Code to oversee the General Visual Examination (PNPS 2.1.8.7) of Primary Containment surfaces every 40 months shall be named by the Engineering Director based on the requirements of the Code.

[7] The Drywell exterior surface of the BWR Mark I containment design is essentially inaccessible to inspection. Additionally, the Drywell interior surface below elevation 9'2" and portions of the vent system exterior surfaces between the Drywell and Torus are also inaccessible. Components or structures shall not be disassembled solely for the purpose of inspection of containment surfaces.

[8] The following areas are exempted from the examination requirements of ASME XI subsection IWE, as allowed by IWE-1220:

(a) Embedded or inaccessible portions of the Drywell that meet the requirements of the original construction Code and of IWE-1232, such as the Drywell shell below elevation 9 foot 2 inches.

(b) Piping, pumps, and valves that are part of the containment system, or which penetrate or are attached to the containment vessel. These components shall be examined in accordance with the rules of IWB or IWC, as appropriate.

5.1.3 Code Category E-C Augmented Examinations

The augmented examinations performed at Pilgrim Station in accordance with IWE-1240 are as listed below. Drawing ISI-IWE-AUG-1 locates the areas requiring examination, and the following text describes the examination extent, method, and acceptance criteria.

[1] Leakage at Annulus Drain Lines, Code Item E4.11

Leakage from the refuel, spent fuel, and equipment pools could lead to corrosion of the Drywell shell by moisture entering the Drywell air gap and/or potentially being entrapped in the sand cushion area below the 9 foot 2 inch elevation. To monitor for this possibility, PNPS has been examining as an augmented examination the annulus drain lines for leakage. This examination is performed each refueling outage after floodup and before draindown of the refuel cavity. Since the drain lines communicate directly with the air gap between the Drywell shell and concrete and is directly above the sand cushion, any water should flow through the drain before possibility wetting the sand cushion. The acceptance for this examination is no detectable leakage from the drain lines.

Examination Required: VT-2 Leakage Test of the Annulus drain lines after each Reactor cavity flood-up and before draindown during each refuel outage.

Refer to PNPS drawings C-71, M-43, and M-41, Memo CSD 90-226, Letter to NRC 87-074 and 91-48, GE RICSIL 009, and PNPS 8.E.19, *"Fuel Pool and Skimmer Surge Tank Instruments."*

[2] Vent Piping, Code Item E4.11

Eight vent pipes from the Drywell elevation 9 ft 2 in. communicate directly with the vent header in the Torus. There is a low point dead leg at the vent pipe to vent header intersection.

It is possible for water to accumulate in the dead legs as a result of workers discharging water into the Torus through the vent. There may also may be some water accumulation due to condensation since the Torus is typically cooler than the Drywell. This situation is exacerbated because some drains were eliminated for structural reasons during the Mark I Torus modification project.

The examination will be a VT-1 of all of eight vent pipes during refuel outage 12. The examination will be performed on the external surface and on the internal surface. The zone of examination is the 1 square foot area at the lowest elevation of each vent pipe. Any water or sludge must be pumped out prior to the examination.

Refer to PNPS drawings CIA51-7 and CIA50-5.

[3] Drywell Shell, Code Item E4.12

Four-inch strips of polyurethane foam filler were specified in drawings to be left in place between the steel Drywell shell and the concrete (air gap) during construction. These strips are horizontal and extend continuously around the Drywell. The strips are specified to be separated by a 5 ft 4 in. center-to-center distance. This construction process extended from elevation 9 ft 2 in. to elevation 90 ft 0 in. These strips are possible sites for the retention of water which may be channeled into the air gap from potential leakage sources including the refuel, spent fuel, and equipment pools.

Since initial augmented inspections performed to Revision 0 of this Procedure determined the refuel and equipment pools have not had leakage resulting in Drywell shell corrosion losses, Revision 1 of this Procedure focuses on the Spent Fuel Pool which has exhibited leakage in the drain system. Examination for this possible corrosion condition will be by ultrasonic thickness measurement of the Drywell shell from the platform at elevation 72 feet. The potential exists for leakage from the bottom of the fuel pool so two strips 6 feet long by 3 inches wide will be examined from the 72 foot elevation proceeding vertically up. The azimuths selected shall be near the fuel pool. In addition to examining for minimum wall thickness values, areas of wall loss from the nominal values will be reported and evaluated. Examinations will be performed every 10 years (Reference QC Inspection Report IR 02-0400).

Refer to PNPS drawings C-118, C-119, M-44, C-112, C-171, C-173 through C-178, M-23, M-413, and M-414.

[4] Torus Shell; Code Item E4.12

The Torus shell interior, downcomers, and vent header were coated with Carbo-Zinc in 1981. Recently, an inspection/coating repair program has been implemented in the Torus due to coating failures. This existing program will continue and additional ultrasonic thickness measurements will be performed to monitor wall thickness.

The highest concentration of visually reported corrosion is in the lower section of the immersed portions of the Torus surface. Ultrasonic thickness measurements will be made at four 1 square foot locations centered 6 feet above the Torus room floor and the furthermost from the Reactor Vessel centerline. Bays 1, 5, 9, and 13 are selected for this sample. Also a 1 square foot section centered on the mean water level (-2 ft 7 in. elevation) will be examined on the same bays. In addition to examining for minimum wall thickness values, any areas of wall loss from the nominal shall be reported and evaluated. These examinations shall be completed each period.

Refer to PNPS drawing C-151.

[5] Refuel Floor Liner Drains, Code Item E4.11

Liner drains for water reservoirs on the refuel floor (e.g., Spent Fuel Pool, Dryer/Separator Pool, and Reactor Cavity) may act as precursors for water leaks which could wet the Drywell shell exterior surface. The drain lines exit to Chemical Radwaste by separate and open drains on elevation 74 feet in the Reactor Building. These drains will be examined for leakage each refuel outage and the results will be reported to Engineering. Leakage will be evaluated by Engineering and further action specified if warranted.

The drains described below will be examined for leakage after cavity floodup. These examinations will be performed by a VT-2 certified person to the extent possible without removing channeling devices.

(a) Spent Fuel Pool Liner - monitoring trench drains
Ref.: C174, C178, M413, and M231 Drain Location:
El. 74' at north wall of Spent Fuel Pool

(b) Dryer Separator Pool Liner - monitoring trench drains
Ref.: C176, C178, M414, and M231
Drain Location: El. 91' at north and south walls of pools

(c) Reactor Cavity
Ref.: C177, M414, and M231

Refer to PNPS drawing ISI-IWE-AUG-2. Refer also to PNPS drawings M-37, M1004 Sheet 86 and Sheet 153, M-3462, M-3496, M-3619, M-3620, M-3623, M-3632, M-3496, C-69, and C-109.

5.2 DRAWINGS

Table 5.2.1 lists the drawings prepared to aid in the performance of the IWE Containment Inspection Program.

TABLE 5.2.1

IWE CONTAINMENT INSPECTION PROGRAM DRAWINGS

Drawing	Title
ISI-IWE-AUG-1	IWE Project Containment Vessel Augmented Inspection Points
ISI-IWE-AUG-2	IWE Boundary Reactor Building Plumbing and Drainage El. 74'3"
ISI-IWE-1	Typical Piping Penetration
ISI-IWE-2	Typical Piping Penetration
ISI-IWE-3	Typical Piping Penetration
ISI-IWE-4	Typical Electrical Penetration
ISI-IWE-5	Typical Electrical Penetration for Coaxial Cable
ISI-IWE-6	Typical Electrical Penetration for Coaxial Cable
ISI-IWE-7	Typical Electrical Penetration for Medium Voltage Power Cable
ISI-IWE-8	Containment Vessel Section
ISI-IWE-9	Drywell Seal and Control Rod Inserts
ISI-IWE-10	CRD Hatch Drywell Penetration X-6
ISI-IWE-11	10'0" Diameter Equipment Door Assembly
ISI-IWE-12	Suppression Chamber Access Penetrations X-200A and X-200B
ISI-IWE-13	Personnel Air Lock

5.3 ASME SECTION XI SUBSECTION IWE INSERVICE INSPECTION TABLES

[1] Table 5.3.1 provides a summary listing of the components for each Examination Category Item No.

[2] Table 5.3.2 provides a complete listing of the IWE components scheduled for examination during the First IWE Inspection Interval.

TABLE 5.3.1

ASME SECTION XI SUBSECTION IWE CONTAINMENT INSERVICE INSPECTION SUMMARY TABLE

Examination Category	Item Number	Description	Number of Components/Areas	Examination Method(s)	Inspection Notes
E-A (Containment Surfaces)	E1.11	Accessible Surface Areas	Drywell, Drywell head, and Torus	General Visual Examination	General Visual Examination shall be performed once each period. Submerged or insulated surfaces are not included within the scope of the General Visual Examination.
	E1.12	Accessible Surface Areas	Drywell, Drywell head, and Torus	VT-3 (Detailed Visual)	Performed at the close of the 10-year inspection interval. Submerged or insulated surfaces shall be examined only to the extent required to achieve coverage of 80% of accessible surfaces.
	E1.20	Vent System	Vent piping, ring header, and downcomer pipes	VT-3 (Detailed Visual)	Performed at the close of the 10-year inspection interval.
E-C (Containment Surfaces Requiring Augmented Examination)	E4.11	Vent Piping	8	VT-1 (2-sided)	Zone of examination is the 1 square foot area at the lowest elevation of each vent pipe.
	E4.11	Annulus Drain Lines	4	VT-2	Performed on four pairs of drains after Reactor cavity floodup and before draindown during each refuel outage.

TABLE 5.3.1 (Cont.)

ASME SECTION XI SUBSECTION IWE
CONTAINMENT INSERVICE INSPECTION SUMMARY TABLE

Examination Category	Item Number	Description	Number of Components/Areas	Examination Method(s)	Inspection Notes
E-C (Containment Surfaces Requiring Augmented Examination)	E4.11	Spent Fuel, Dryer/ Separator Pool, and Reactor Cavity	4	VT-2	Performed on drain locations on the Reactor Building 74' elevation once each period while flooded up.
	E4.12	Upper Drywell Shell	2	UT	An area 6 ft tall by 3 in. wide shall be examined at two locations (azimuths 252 and 288 degrees) between the 72' and 77' elevations adjacent to the Spent Fuel Pool. Examinations shall be performed once every 10 years (Reference QC Inspection Report IR 02-0400).

TABLE 5.3.1 (Cont.)

ASME SECTION XI SUBSECTION IWE CONTAINMENT INSERVICE INSPECTION SUMMARY TABLE

Examination Category	Item Number	Description	Number of Components/Areas	Examination Method(s)	Inspection Notes
E-C (Containment Surfaces Requiring Augmented Examination)	E4.12	Torus Shell	8	UT	Test areas will be 1 square foot locations centered 6 feet above the Torus Room floor and furthermost from the Reactor Vessel centerline in Torus Room bays 1, 5, 9, and 13. Additionally, a 1 square foot section centered on the Torus mean water level (elev. -2 ft 7 in.) will be examined in the same bays. Examinations shall be performed once each period.
E-D (Seals, Gaskets, Moisture Barriers)	E5.10	Seals	26 electrical penetrations	VT-3	PRR-E1
	E5.20	Gaskets	35	VT-3	PRR-E1
	E5.30	Moisture Barriers	0	VT-3	Interior moisture barrier located at Drywell elevation 9 feet between Drywell shell and concrete floor shall be examined once per interval. During initial examination (RFO #12), it was determined a barrier does not exist nor is one required at 9 foot elevation. Exterior moisture barrier above the Drywell sand cushion area is inaccessible.
E-G (Pressure-Retaining Bolting)	E8.10	Bolted Connections	35	VT-1	100% of components to be examined during the 10-year interval.
	E8.20	Bolted Connections	35	Bolt torque or tension test	PRR-E4

TABLE 5.3.1 (Cont.)

ASME SECTION XI SUBSECTION IWE CONTAINMENT INSERVICE INSPECTION SUMMARY TABLE

Examination Category	Item Number	Description	Number of Components/Areas	Examination Method(s)	Inspection Notes
E-P (All pressure retaining components)	E9.20	Containment Penetration Bellows	24	Appendix J Type B test	24 penetrations with bellows
	E9.30	Air locks	1	Appendix J Type B test	Personnel Hatch
	E9.40	Seals and Gaskets	63	Appendix J Type B test	28 electrical penetrations with seals and 35 penetrations with gaskets
F-A (Class MC supports)	F1.40B	Torus saddle supports	16	VT-3	25% of saddle supports examined during the inspection interval
	F1.40B	Torus earthquake ties	4	VT-3	100% of Torus earthquake tie supports examined during the inspection interval
	F1.40C	Drywell Stabilizers	8	VT-3	25% of supports examined during the inspection interval

TABLE 5.3.2

PILGRIM NUCLEAR POWER STATION IWE COMPONENTS SCHEDULED FOR EXAMINATION DURING 1st IWE INTERVAL

Component	Description	Code Cate-gory	Code Item	IWE Period	ISI Class	Sys-tem	Location
IWE-GVWD-01	General Visual Walkdown	E-A	E1.11	1,2	MC	Cont	Various
IWE-DV-01	Detailed Visual	E-A	E1.12	2	MC	Cont	Various
IWE-VS-01	Vent System	E-A	E1.20	3	MC	Cont	Torus
IWE-ANNDRN-080	Annulus Drains (2) at 80 AZ	E-C	E4.11	1,2,3	MC	Cont	Torus Room
IWE-ANNDRN-170	Annulus Drains (2) at 170 AZ	E-C	E4.11	1,2,3	MC	Cont	Torus Room
IWE-ANNDRN-260	Annulus Drains (2) at 260 AZ	E-C	E4.11	1,2,3	MC	Cont	Torus Room
IWE-ANNDRN-350	Annulus Drains (2) at 350 AZ	E-C	E4.11	1,2,3	MC	Cont	Torus Room
IWE-LINERDRAINS	Liner Drains	E-C	E4.11	1,2,3	MC	Cont	RB 74'
IWE-VENT-022	Augmented Vent Pipe at 22 AZ	E-C	E4.11	1,2,3	MC	Cont	Torus
IWE-VENT-067	Augmented Vent Pipe at 67 AZ	E-C	E4.11	1,2,3	MC	Cont	Torus
IWE-Vent-112	Augmented Vent Pipe at 112 AZ	E-C	E4.11	1,2,3	MC	Cont	Torus
IWE-VENT-157	Augmented Vent Pipe at 157 AZ	E-C	E4.11	1,2,3	MC	Cont	Torus
IWE-VENT-202	Augmented Vent Pipe at 200 AZ	E-C	E4.11	1,2,3	MC	Cont	Torus
IWE-VENT-247	Augmented Vent Pipe at 247 AZ	E-C	E4.11	1,2,3	MC	Cont	Torus
IWE-VENT-292	Augmented Vent Pipe at 292 AZ	E-C	E4.11	1,2,3	MC	Cont	Torus
IWE-VENT-337	Augmented Vent Pipe at 337 AZ	E-C	E4.11	1,2,3	MC	Cont	Torus
IWE-SNDCUSH-035	Augmented Drywell UT at 9 ft 035 AZ	E-C	E4.12	1	MC	Cont	Drywell
IWE-SNDCUSH-125	Augmented Drywell UT at 9 ft 125 AZ	E-C	E4.12	1	MC	Cont	Drywell
IWE-SNDCUSH-215	Augmented Drywell UT at 9 ft 215 AZ	E-C	E4.12	1	MC	Cont	Drywell
IWE-SNDCUSH-305	Augmented Drywell UT at 9 ft 305 AZ	E-C	E4.12	1	MC	Cont	Drywell
IWE-TORUS-LOWER-B1	Augmented Torus UT elev. -11 ft 6 in Bay 1	E-C	E4.12	1,2,3	MC	Cont	Torus Room
IWE-TORUS-LOWER-B13	Augmented Torus UT elev. -11 ft 6 in Bay 13	E-C	E4.12	1,2,3	MC	Cont	Torus Room
IWE-TORUS-LOWER-B5	Augmented Torus UT elev. -11 ft 6 in Bay 5	E-C	E4.12	1,2,3	MC	Cont	Torus Room
IWE-TORUS-LOWER-B9	Augmented Torus UT elev. -11 ft 6 in Bay 9	E-C	E4.12	1,2,3	MC	Cont	Torus Room
IWE-TORUS-MWL-B1	Augmented Torus UT at MWL Bay 1	E-C	E4.12	1,2,3	MC	Cont	Torus Room

TABLE 5.3.2 (Cont.)

PILGRIM NUCLEAR POWER STATION IWE COMPONENTS SCHEDULED FOR EXAMINATION DURING 1st IWE INTERVAL

Component	Description	Code Cate-gory	Code Item	IWE Period	ISI Class	Sys-tem	Location
IWE-TORUS-MWL-B13	Augmented Torus UT at MWL Bay 13	E-C	E4.12	1,2,3	MC	Cont	Torus Room
IWE-TORUS-MWL-B5	Augmented Torus UT at MWL Bay 5	E-C	E4.12	1,2,3	MC	Cont	Torus Room
IWE-TORUS-MWL-B9	Augmented Torus UT at MWL Bay 9	E-C	E4.12	1,2,3	MC	Cont	Torus Room
IWE-UPDW-72-252	Augmented Drywell UT at 72 ft 252 AZ	E-C	E4.12	1	MC	Cont	Drywell
IWE-UPDW-72-288	Augmented Drywell UT at 72 ft 288 AZ	E-C	E4.12	1	MC	Cont	Drywell
IWE-UPDW-83-072	Augmented Drywell UT at 83 ft 72 AZ	E-C	E4.12	1	MC	Cont	Drywell
IWE-UPDW-83-108	Augmented Drywell UT at 83 ft 108 AZ	E-C	E4.12	1	MC	Cont	Drywell
IWE-UPDW-83-252	Augmented Drywell UT at 83 ft 252 AZ	E-C	E4.12	1	MC	Cont	Drywell
IWE-UPDW-83-288	Augmented Drywell UT at 83 ft 288 AZ	E-C	E4.12	1	MC	Cont	Drywell
IWE-CB-DWHEAD	Containment Bolting	E-G	E8.10	1	MC	Cont	RB 117'
IWE-CB-GIBS270	Containment Bolting	E-G	E8.10	1	MC	Cont	Drywell
IWE-CB-X200B	Containment Bolting	E-G	E8.10	1	MC	Cont	Torus Room
IWE-CB-X203A	Containment Bolting	E-G	E8.10	1	MC	Cont	Torus Interior
IWE-CB-X203B	Containment Bolting	E-G	E8.10	1	MC	Cont	Torus Interior
IWE-CB-X203C	Containment Bolting	E-G	E8.10	1	MC	Cont	Torus Interior
IWE-CB-X213A	Containment Bolting	E-G	E8.10	1	MC	Cont	Torus Room
IWE-CB-X35A	Containment Bolting	E-G	E8.10	1	MC	Cont	TIP Room
IWE-CB-X4	Containment Bolting	E-G	E8.10	1	MC	Cont	RB 117'
IWE-CB-X6	Containment Bolting	E-G	E8.10	1	MC	Cont	RB 23'
IWE-CB-GIBS135	Containment Bolting	E-G	E8.10	2	MC	Cont	Drywell
IWE-CB-GIBS180	Containment Bolting	E-G	E8.10	2	MC	Cont	Drywell
IWE-CB-GIBS225	Containment Bolting	E-G	E8.10	2	MC	Cont	Drywell
IWE-CB-GIBS315	Containment Bolting	E-G	E8.10	2	MC	Cont	Drywell

TABLE 5.3.2 (Cont.)

PILGRIM NUCLEAR POWER STATION IWE COMPONENTS SCHEDULED FOR EXAMINATION DURING 1st IWE INTERVAL

Component	Description	Code Cate-gory	Code Item	IWE Period	ISI Class	Sys-tem	Location
IWE-CB-X1	Containment Bolting	E-G	E8.10	2	MC	Cont	Drywell
IWE-CB-X200A	Containment Bolting	E-G	E8.10	2	MC	Cont	Torus Room
IWE-CB-X203D	Containment Bolting	E-G	E8.10	2	MC	Cont	Torus Interior
IWE-CB-X203E	Containment Bolting	E-G	E8.10	2	MC	Cont	Torus Interior
IWE-CB-X203F	Containment Bolting	E-G	E8.10	2	MC	Cont	Torus Interior
IWE-CB-X213B	Containment Bolting	E-G	E8.10	2	MC	Cont	Torus Room
IWE-CB-X35B	Containment Bolting	E-G	E8.10	2	MC	Cont	TIP Room
IWE-CB-X35C	Containment Bolting	E-G	E8.10	2	MC	Cont	TIP Room
IWE-CB-GIBS360	Containment Bolting	E-G	E8.10	3	MC	Cont	Drywell
IWE-CB-GIBS45	Containment Bolting	E-G	E8.10	3	MC	Cont	Drywell
IWE-CB-GIBS90	Containment Bolting	E-G	E8.10	3	MC	Cont	Drywell
IWE-CB-X2	Containment Bolting	E-G	E8.10	3	MC	Cont	Drywell
IWE-CB-X203G	Containment Bolting	E-G	E8.10	3	MC	Cont	Torus Interior
IWE-CB-X203H	Containment Bolting	E-G	E8.10	3	MC	Cont	Torus Interior
IWE-CB-X203J	Containment Bolting	E-G	E8.10	3	MC	Cont	Torus Interior
IWE-CB-X203K	Containment Bolting	E-G	E8.10	3	MC	Cont	Torus Interior
IWE-CB-X230	Containment Bolting	E-G	E8.10	3	MC	Cont	Torus Room
IWE-CB-X35D	Containment Bolting	E-G	E8.10	3	MC	Cont	TIP Room
IWE-CB-X35E	Containment Bolting	E-G	E8.10	3	MC	Cont	TIP Room
IWE-CB-X43	Containment Bolting	E-G	E8.10	3	MC	Cont	B RHR VV Room
IWE-CB-X47	Containment Bolting	E-G	E8.10	3	MC	Cont	Steam Tunnel
H-50-1-TORUS BAY 13	Torus Supports	F-A	F1.40B	1	MC	Cont	Torus Room
H-50-1-TORUS BAY 9	Torus Supports	F-A	F1.40B	1	MC	Cont	Torus Room
H-50-1-TORUS BAY 1	Torus Supports	F-A	F1.40B	2	MC	Cont	Torus Room
H-50-1-TORUS BAY 5	Torus Supports	F-A	F1.40B	2	MC	Cont	Torus Room
H-50-1-270GIBS	Drywell Stabilizer	F-A	F1.40C	1	MC	Cont	Drywell
H-50-1-315GIBS	Drywell Stabilizer	F-A	F1.40C	2	MC	Cont	Drywell

5.4 IWE CONTAINMENT INSPECTION RELIEF REQUESTS

5.4.1 IWE Relief Request Index

TABLE 5.4.1

IWE CONTAINMENT INSPECTION PROGRAM
RELIEF REQUEST INDEX

Relief Request	Rev.	Date	Relief Description
PRR-E1	0	11/23/98	Examination of Seals and Gaskets
PRR-E2	0	11/23/98	Alternative Provisions for Qualification of NDE Personnel
PRR-E3	0	11/23/98	Successive Examinations for Components Found Acceptable for Continued Service
PRR-E4	0	11/23/98	Alternative Provisions for Pressure-Retaining Bolting Examinations
PRR-E5	0	11/23/98	Alternative Provisions for Visual Examination of Coatings Prior to Removal
PRR-E6	0	11/23/98	Alternative Provisions for Preservice Examinations of New Coatings

5.4.2 IWE Relief Request Number PRR-E1

RELIEF REQUEST NUMBER PRR-E1
Revision 0

COMPONENT IDENTIFICATION

Seals and gaskets of Class MC pressure retaining components, Examination Category E-D, Item Numbers E5.10 and E5.20 of IWE-2500, "Examination and Pressure Test Requirements," Table IWE-2500-1, ASME Section XI, 1992 Edition, 1992 Addenda.

CODE REQUIREMENT

IWE-2500, Table IWE-2500-1 requires seals and gaskets on air locks, hatches, and other devices to be visually examined, VT-3, once each interval to assure containment leak-tight integrity. Relief is requested from performing the Code-required visual examination, VT-3, on the above identified metal containment seals and gaskets in accordance with 10CFR50.55a(a)(3)(ii).

BASIS FOR RELIEF

10CFR50.55a was amended, as cited in the Federal Register (61FR41303), to require the use of the 1992 Edition, 1992 Addenda, of Section XI when performing containment examinations. The penetrations discussed below contain seals and gaskets:

Electrical Penetrations

Electrical penetrations include electrical power, signal, and instrument leads with the penetrating sleeves welded to the Primary Containment vessel. Medium voltage (600V and 5kV) power penetrations at Pilgrim Station have primary seals made of alumina-ceramic materials. The low voltage power control and instrumentation cable and coaxial cable penetrations use a bonding resin to maintain the leak-tight integrity of the containment penetrating sleeves. Each penetration is pressurized to 45 psig with dry nitrogen to maintain and monitor integrity and to prevent the intrusion of moisture into the penetration.

These seals and gaskets cannot be inspected without disassembly of the penetration to gain access to the seals and gaskets.

Drywell Head, Drywell Head Manway, Drywell Personnel and Equipment Hatches, CRD Service, Torus Access and Drywell Stabilizer Access Hatches

The personnel hatch utilizes an inner and outer door with gasket surfaces to ensure leak-tight integrity. This hatch also contains other gaskets and seals such as handwheel shaft seals, electrical penetrations, blank flanges, and equalizing pressure connections which require disassembly to gain access to the gaskets and seals.

The other hatches listed above utilize seals and/or gaskets in Appendix J testable joints to maintain leak-tight integrity. Seals and gaskets receive a 10CFR50 Appendix J Type B test. As noted in 10CFR50 Appendix J, the purpose of Type B tests is to measure leakage of containment or penetrations whose design incorporates resilient seals, gaskets, sealant compounds, and electrical penetrations fitted with flexible metal seal assemblies. Examination of seals and gaskets require the joints, which are proven adequate through Appendix J testing, be disassembled. For electrical penetrations, this would involve a premaintenance Appendix J test, determination of cables at electrical penetrations if enough cable slack is not available, disassembly of the joint, removal and examination of the seals and gaskets, reassembly of the joint, retermination of the cables if necessary, postmaintenance testing of the cables, and a postmaintenance Appendix J test of the penetration. The work required for containment hatches and other bolted joints would be similar except for the de-termination, retermination, and testing of cables. This imposes the risk that equipment could be damaged. The 1992 Edition, 1993 Addenda, of ASME Section XI recognizes that disassembly of joints to perform these examinations is not warranted. Note 1 in Examination Category E-D was modified in the 1995 Edition of ASME Section XI to state that sealed or gasket connections need not be disassembled solely for performance of examinations. However, without disassembly, most of the surface of the seals and gaskets would be inaccessible.

For those penetrations that are routinely disassembled, a Type B test is required upon final assembly and prior to startup. Since the Type B test will assure the leak-tight integrity of Primary Containment, the performance of the visual examination would not increase the level of safety or quality.

Seals and gaskets are not part of the containment pressure boundary under current Code rules NE-1220(b). When the air locks and hatches containing these materials are tested in accordance with 10CFR50, Appendix J, degradation of the seal or gasket material would be revealed by an increase in the leakage rate. Corrective measures would be applied and the component retested. Repair or replacement of seals and gaskets is not subject to Code (1992 Edition, 1992 Addenda) rules in accordance with Paragraph IWA-4111(b)(5) of ASME Section XI.

The visual examination of seals and gaskets in accordance with IWE-2500, Table IWE-2500-1, is a burden without any compensating increase in the level of safety or quality.

Relief is requested from performing the Code-required visual examination, VT-3, on the above identified metal containment seals and gaskets in accordance with 10CFR50.55a(a)(3)(ii). Compliance with the specified requirements of this section would result in hardship or unusual difficulty without a compensating increase in the level of quality and safety. Testing the seals and gaskets in accordance with 10CFR50 Appendix J will provide adequate assurance of the leak-tight integrity of the seals and gaskets.

The requirement to examine seals and gaskets has been removed in the rewrite of Subsection IWE of ASME Section XI which has been approved by ASME and was published in 1998.

PROPOSED ALTERNATIVE EXAMINATIONS

The leak-tightness of seals and gaskets will be tested in accordance with 10CFR50 Appendix J. The 10CFR50 Appendix J Type B testing is performed at least once each inspection interval.

APPLICABLE TIME PERIOD

Relief is requested for the first ten-year interval of the Pilgrim Station IWE Containment Inspection Program, beginning September 6, 1998.

5.4.3 IWE Relief Request Number PRR-E2

RELIEF REQUEST NUMBER PRR-E2
Revision 0

COMPONENT IDENTIFICATION

All components subject to examination in accordance with Subsection IWE of the 1992 Edition, 1992 Addenda of ASME Section XI.

CODE REQUIREMENT

Subarticle IWA-2300, "Qualification of Nondestructive Examination Personnel," requires qualification of nondestructive examination personnel to CP-189-1991, "Standard for Qualification and Certification of Nondestructive Testing Personnel," as amended by the ASME Section XI.

BASIS FOR RELIEF

10CFR50.55a was amended, as cited in the Federal Register (61FR41303), to require the use of the 1992 Edition, 1992 Addenda, of Section XI, when performing containment examinations. In addition to the requirements of Subsection IWE, this also imposes the requirements of Subsection IWA, General Requirements, of the 1992 Edition, 1992 Addenda of Section XI. Subarticle IWA-2300 requires qualification of nondestructive examination personnel to CP-189, as amended by Subarticle IWA-2300.

A written practice based on the requirements of CP-189, as amended by the requirements of the Subarticle IWA-2300, to implement Subsection IWE duplicates efforts already in place for all other subsections. The Pilgrim Nuclear Power Station Third Ten-Year Inservice Inspection Program is written to meet the 1989 Edition of Section XI. Subarticle IWA-2300 of this edition requires a written practice based on SNT-TC-1A, "Personnel Qualification and Certification in Nondestructive Testing," as amended by the requirements of Subarticle IWA-2300. Further, Subarticle IWA-2300 of the 1992 Edition, 1992 Addenda, states, "Certifications based on SNT-TC-1A are valid until recertification is required."

Visual examination is the primary nondestructive examination method required by Subsection IWE. Neither CP-189 nor SNT-TC-1A specifically includes visual examination. Therefore, the Code requires qualification and certification to comparable levels as defined in CP-189 or SNT-TC-1A, as applicable, and the employer's written practice. Ultrasonic thickness examinations may also be required by Table IWE-2500-1. These examinations are relatively simple and do not require an extensive training and qualification program. Therefore, use of CP-189 in place of SNT-TC-1A will not improve the capability of examination personnel to perform the visual and ultrasonic thickness examinations required by IWE.

Development and administration of a second program would not enhance safety or quality and would serve as a burden, particularly in developing a second written practice, tracking of certifications, and duplication of paperwork. This duplication would also apply to nondestructive examination (NDE) vendor programs. Updating to the 1992 Edition, 1992 Addenda, for Subsections IWB, IWC, etc., would require a similar request for relief.

Relief is requested from the provisions of Subarticle IWA-2300, "Qualification of Nondestructive Examination Personnel in accordance with 10CFR50.55a(a)(3)(ii). Compliance with the specified requirements of this section would result in hardship or unusual difficulty without a compensating increase in the level of quality and safety.

The requirement to comply with IWA-2300 has been removed in the rewrite of Subsection IWE of ASME Section XI. This rewrite has been approved by ASME and was published in 1998.

PROPOSED ALTERNATIVE EXAMINATIONS

Examinations required by Subsections IWE shall be conducted by personnel qualified and certified to a written practice based on SNT-TC-1A and the 1989 Edition of ASME Section XI. Visual examination personnel will receive specific training in conducting containment examinations.

APPLICABLE TIME PERIOD

Relief is requested for the first ten-year interval of the Pilgrim Station IWE Containment Inspection Program, beginning September 6, 1998.

5.4.4 IWE Relief Request Number PRR-E3

RELIEF REQUEST NUMBER PRR-E3
Revision 0

COMPONENT IDENTIFICATION

All Class MC, Paragraphs IWE-2420(b) and IWE-2420(c) successive examination requirements for components found acceptable for continued service.

CODE REQUIREMENT

Paragraphs IWE-2420(b) and IWE-2420(c) of the 1992 Edition, 1992 Addenda of ASME Section XI, require that when component examination results require evaluation of flaws, evaluation of areas of degradation, or repairs in accordance with Article IWE-3000, "Acceptance Standards," and the component is found to be acceptable for continued service, the areas containing such flaws, degradation, or repairs shall be reexamined during the next inspection period listed in the schedule of the inspection program of Paragraph IWE-2411, "Inspection Program A," or Paragraph IWE-2412, "Inspection Program B," in accordance with Table IWE-2500-1, Examination Category E-C. Relief is requested from the requirement of Paragraphs IWE-2420(b) and IWE-2420(c) to perform successive examination of repairs.

BASIS FOR RELIEF

10CFR50.55a was amended, as cited in the Federal Register (61FR41303), to require the use of the 1992 Edition, 1992 Addenda, of Section XI, when performing containment examinations. The purpose of a repair is to restore the component to an acceptable condition for continued service in accordance with the acceptance standards of Article IWE-3000. Paragraph IWA-4150, "Verification of Acceptability," requires the owner to conduct an evaluation of the suitability of the repair including consideration of the cause of failure.

If the repair has restored the component to an acceptable condition, successive examinations are not warranted. If the repair was not suitable, then the repair does not meet Code requirements and the component is not acceptable for continued service. Neither Paragraph IWB-2420(b), Paragraph IWC-2420(b), nor Paragraph IWD-2420(b) requires a repair to be subject to successive examination requirements. Furthermore, if the repair area is subject to accelerated degradation, it would still require augmented examination in accordance with Table IWE-2500-1, Examination Category E-C.

The successive examination of repairs in accordance with Paragraphs IWE-2420(b) and IWE-2420(c) constitutes a burden without a compensating increase in quality or safety.

Relief is requested in accordance with 10CFR50.55a(a)(3)(ii). Compliance with the specified requirements of this section would result in hardship or unusual difficulty without a compensating increase in the level of quality and safety.

The requirement to perform successive examinations following repairs has been removed in the rewrite of Subsection IWE of ASME Section XI. This rewrite has been approved by ASME and was published in 1998.

PROPOSED ALTERNATIVE EXAMINATIONS

No alternative examinations are proposed as successive examinations in accordance with Paragraphs IWE-2420(b) and IWE-2420(c) are not required for repairs made in accordance with Article IWA-4000.

APPLICABLE TIME PERIOD

Relief is requested for the first ten-year interval of the Pilgrim Station IWE Containment Inspection Program, beginning September 6, 1998.

5.4.5 IWE Relief Request Number PRR-E4

RELIEF REQUEST NUMBER PRR-E4

Revision 0

COMPONENT IDENTIFICATION

Class MC pressure retaining bolting.

CODE REQUIREMENT

ASME Section XI, 1992 Edition with the 1992 Addenda, Table IWE-2500-1, Examination Category E-G, Pressure Retaining Bolting, Item E8.20. Relief is requested from ASME Section XI 1992 Edition, 1992 Addenda, Table IWE-2500-1 Examination Category E-G, Pressure Retaining Bolting, Item E8.20. Table IWE-2500-1 requires a bolt torque or tension test on bolted connections that have not been disassembled and reassembled during the inspection interval.

BASIS FOR RELIEF

10CFR50.55a was amended, as cited in the Federal Register (61FR41303), to require the use of the 1992 Edition, 1992 Addenda, of ASME Section XI when performing containment examinations. Bolt torque or tension testing is required on bolted connections that have not been disassembled and reassembled during the inspection interval. Determination of the torque or tension value would require that the bolting be untorqued and then retorqued or retensioned.

Each containment penetration receives a 10CFR50 Appendix J Type B test in accordance with the specified testing frequencies. As noted in 10CFR50 Appendix J, the purpose of Type B tests is to measure leakage of containment penetrations whose design incorporates resilient seals, gaskets, sealant compounds, and electrical penetrations fitted with flexible metal seal assemblies. The performance of the Type B test itself proves that the bolt torque or tension remains adequate to provide a leak rate that is within acceptable limits. The torque or tension value of bolting only becomes an issue if the leak rate is excessive. Once a bolt is torqued or tensioned, it is not subject to dynamic loading that could cause it to experience significant change. Appendix J testing and visual inspection are adequate to demonstrate that the design function is met. Torque or tension testing is not required for any other ASME Section XI, Class 1, 2, or 3 bolted connections or their supports as part of the inservice inspection program.

Relief is requested in accordance with 10CFR50.55a(a)(3)(ii). Untorquing and subsequent retorquing (or other torque testing methods) of bolted connections which are verified not to experience unacceptable leakage through 10CFR50 Appendix J Type B testing results in hardship or unusual difficulty without a compensating increase in the level of quality and safety.

The requirement to perform bolt torque or tension tests has been removed in the rewrite of Subsection IWE of ASME Section XI which has been approved by ASME and was published in 1998.

PROPOSED ALTERNATE EXAMINATION(S)

The following examinations and tests required by Subsection IWE ensure the structural integrity and the leak-tightness of Class MC pressure retaining bolting and, therefore, no additional alternative examinations are proposed:

- Exposed surfaces of bolted connections shall be visually examined in accordance with requirements of Table IWE-2500-1, Examination Category E-G, Pressure Retaining Bolting, Item No. E8.10; and

- Bolted connections shall meet the pressure test requirements of Table IWE-2500-1, Examination Category E-P, All Pressure Retaining Components, Item E9.40.

APPLICABLE TIME PERIOD

Relief is requested for the first ten-year interval of the Pilgrim Station IWE Containment Inspection Program, beginning September 6, 1998.

5.4.6 IWE Relief Request Number PRR-E5

RELIEF REQUEST NUMBER PRR-E5

Revision 0

COMPONENT IDENTIFICATION

All Class MC, Subarticle IWE-2500(b) visual examinations in accordance with Table IWE-2500-1 of painted or coated containment components prior to removal of paint or coatings.

CODE REQUIREMENT(S)

ASME Section XI, 1992 Edition, 1992 Addenda, Subarticle IWE-2500(b) requires that when paint or coatings are to be removed, the paint or coatings shall be visually examined in accordance with Table IWE-2500-1 prior to removal.

BASIS FOR RELIEF

10CFR50.55a was amended, as cited in the Federal Register (61FR41303), to require the use of the 1992 Edition, 1992 Addenda, of ASME Section XI when performing containment examinations. Paint and coatings are not part of the containment pressure boundary under current Code rules as they are not associated with the pressure retaining function of the component (Paragraph NE-2110 (b)(5) of ASME Section III). The containment interior surfaces at Pilgrim Station are painted to prevent rusting and are exposed to an inert atmosphere at all times except during refuel or maintenance outages. The exterior surfaces of the Torus, vent system, and Drywell head are also painted and exist in a controlled atmosphere (Secondary Containment). Neither paint nor coatings contributes to the structural integrity or leak-tightness of the containment. Furthermore, the paint and coatings on the containment pressure boundary were not subject to Code rules when they were originally applied and are not subject to ASME Section XI rules for repair or replacement in accordance with IWA-4111(b)(5). Degradation or discoloration of the paint or coating materials on containment would be an indicator of potential degradation of the containment pressure boundary. Additional measures would have to be employed to determine the nature and extent of any degradation, if present. The application of ASME Section XI rules for removal of paint or coatings when unrelated to an ASME Section XI repair or replacement activity is a burden without a compensating increase in quality or safety.

Relief is requested in accordance with 10CFR50.55a(a)(3)(i). PNPS Specifications C-98A, M530, and M531 currently control containment coating activities at PNPS and provide an adequate level of quality and safety as they conform to Regulatory Guide 1.54 and ANSI Standards N101.4 and N5.12. All containment coating work at PNPS is performed by qualified vendors approved to provide coating services subject to 10CFR50 Appendix B controls on Special Processes. Additionally, the General Visual Walkdown required by subsection IWE to be performed once every inspection period will provide an adequate periodic assessment of the condition of containment coatings.

The requirement to inspect coatings prior to removal has been removed in the rewrite of Subsection IWE of ASME Section XI. This rewrite has been approved by ASME and was published in 1998.

PROPOSED ALTERNATIVE EXAMINATIONS

The condition of the containment vessel base material will be verified prior to the application of new paint or coating as required by PNPS Specifications C-98A, M530, and M531. If degradation is identified, additional measures will be applied to determine whether the containment pressure boundary is affected. Repairs to the primary containment boundary, if required, will be conducted in accordance with ASME Section XI Code rules.

APPLICABLE TIME PERIOD

Relief is requested for the first ten-year interval of the Pilgrim Station IWE Containment Inspection Program, beginning September 6, 1998.

5.4.7 IWE Relief Request Number PRR-E6

RELIEF REQUEST NUMBER PRR-E6

Revision 0

COMPONENT IDENTIFICATION

All Class MC, Subarticle IWE-2200(g), preservice examination requirements of reapplied painted or coated containments.

CODE REQUIREMENT

ASME Section XI, 1992 Edition, 1992 Addenda, Subsection IWE-2200(g) requires that when paint or coatings are reapplied, the condition of the new paint or coating shall be documented in the preservice examination records. Relief is requested from the requirement to perform a preservice inspection of new paint or coatings.

BASIS FOR RELIEF

Paint and coatings are not part of the containment pressure boundary under current Code rules as they are not associated with the pressure retaining function of the component (Paragraph NE-2110 (b)(5) of ASME Section III). Neither paint nor coatings contributes to the structural integrity or leak-tightness of the containment. Furthermore, the paint and coatings on the containment pressure boundary were not subject to Code rules when they were originally applied and are not subject to ASME Section XI rules for repair or replacement in accordance with IWA-4111(b)(5). The adequacy of applied coatings is verified through the inspections and tests performed by qualified vendors approved by Entergy to provide coating services at PNPS subject to 10CFR50 Appendix B controls on Special Processes. Primary Containment coating activities at PNPS are currently controlled by PNPS Specifications C-98A, M530, and M531 which conform to Regulatory Guide 1.54 and ANSI Standards N101.4 and N5.12. Additionally, the General Visual Walkdown required by subsection IWE to be performed once each inspection period will provide an adequate periodic assessment of the condition of containment coatings.

Recording the condition of reapplied coating in the preservice record does not substantiate the containment structural integrity. Should deterioration of the coating in the reapplied area occur, the area will require additional evaluation regardless of the preservice record. Recording the condition of new paint or coating in the preservice records does not increase the level of quality and safety of the containment.

In SECY 96-080, "Issuance of Final Amendment to 10CFR Section 50.55a to Incorporate by Reference the ASME Boiler and Pressure Vessel Code (ASME Code), Section XI, Division 1, Subsection IWE and Subsection IWL," dated April 17, 1996, response to Comment 3.2 about IWE-2200(g) states, "In the NRC's opinion, this does not mean that a visual examination must be performed with every application of paint or coating. A visual examination of the topcoat to determine the soundness and the condition of the topcoat should be sufficient." This is currently accomplished through the inspections required by Specifications C-98A, M530, and M531 and performed by qualified vendors approved to provide coating services at PNPS subject to 10CFR50 Appendix B controls.

Relief is requested in accordance with 10CFR50.55a(a)(3)(i). The inspections and tests performed in accordance with PNPS Specifications C-98A, M530, and M531 provide an adequate level of quality and safety since the specifications conform to Regulatory Guide 1.54 and ANSI Standards N101.4 and N5.12. The requirement to perform a preservice examination when paint or coatings are reapplied has been removed in the rewrite of Subsection IWE of ASME Section XI. This rewrite has been approved by ASME and is scheduled to be published in 1998.

PROPOSED ALTERNATE EXAMINATIONS

Reapplied paint and coatings on the containment vessel will be examined in accordance with the requirements of PNPS Specifications C-98A, M530, and M531. Although repairs to paint or coatings are not subject to the repair/replacement rules of ASME XI (Inquiry 97-22), repairs to the Primary Containment boundary, if required, will be conducted in accordance with ASME Section XI Code rules.

APPLICABLE TIME PERIOD

Relief is requested for the first ten-year interval of the Pilgrim Station IWE Containment Inspection Program, beginning September 6, 1998.

5.5 NRC CORRESPONDENCE

Edison letter 2.98.151, dated November 23, 1998, to the NRC: Request for Relief from the 1992 ed. with 1992 add. of ASME XI, Subsection IWE.

6.0 PROCEDURE

[1] Condition Reports and Nonconformance Reports shall require the following for close-out and shall be stated in the originator's request:

- Condition that leads to degradation.
- Acceptability of each flaw or area.
- Need for additional examinations to verify that similar degradation does not exist in similar components.
- Description of necessary corrective action.
- Number and type of additional examinations to ensure detection of similar degradation in similar components.

[2] The scheduling of the General Visual Walkdown shall be coordinated with people responsible for Appendix J testing and shall be performed each inspection period. The General Visual Walkdown shall coincide with the Appendix J containment walkdowns.

[3] Quality Assurance ISI personnel shall obtain the services of a responsible engineer to oversee the General Visual Walkdown. The person will be provided by Engineering by memo or equivalent.

7.0 RECORDS

None

8.0 ATTACHMENTS

None