Entergy

Entergy Nuclear Northeast
Entergy Nuclear Operations, Inc.
James A. Fitzpatrick NPP
P.O. Box 110
Lycoming, NY 13093
Tel 315 349 6024 Fax 315 349 6480

T.A. Sullivan
Site Vice President - JAF

July 28, 2003
JAFP-03-0108

U. S. Nuclear Regulatory Commission
Attn: Document Control Desk
Mail Stop O-P1-17
Washington, DC 20555-0001

SUBJECT: James A. FitzPatrick Nuclear Power Plant
Docket No. 50-333
Proposed License Amendment to Provide a One-time Integrated Leak Rate Test (ILRT) Interval Extension

Gentlemen:

In accordance with the provisions of 10 CFR 50.90, Entergy Nuclear Operations, Inc. (ENO) is submitting a request for an amendment to the Technical Specifications (TS) for the James A. FitzPatrick Nuclear Power Plant (JAFNPP).

The proposed license amendment would revise Technical Specification section 5.5.6 "Primary Containment Leakage Rate Testing Program" to allow a one-time interval extension for the JAF Type A, Integrated Leakage Rate Test (ILRT) of no more than five (5) years. This revision takes a one time exception to the ten (10) year frequency of the performance-based leakage rate testing program for Type A tests as required by NEI 94-01, revision 0, "Industry Guideline For Implementing Performance-Based Option of 10 CFR Part 50, Appendix J", and endorsed by 10 CFR 50, Appendix J, Option B. The one time exception is to the requirement of NEI 94-01 to perform an integrated leak rate test (ILRT) at a frequency of up to ten years, with allowance for a 15 month extension. The exception is to allow ILRT testing within fifteen years from the last ILRT, performed on March 7, 1995. This application represents a cost beneficial licensing change. The integrated leak rate test imposes significant expense on the station while the safety benefit of performing it within 10 years, versus 15 years, is minimal.

This request is made with a risk-informed basis as described in Regulatory Guide 1.177, "An Approach for Plant-Specific, Risk-Informed Decisionmaking: Technical Specifications." The plant specific risk assessment evaluation performed in support of this request, Engineering Report JAF-RPT-03-00007, "Risk Impact Assessment of Extending Containment Type A Test Interval" is forwarded as Attachment 4. This 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

A017

Contact 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 extension submittal. Detailed bases and considerations for the risk assessment evaluation are identified in Attachment 4.

The conclusion of the risk assessment is that the one-time increase in the Type A test interval from ten to fifteen years would result in negligible effect on plant risk. In particular, the conclusion of the plant internal events risk associated with extending the Type A ILRT interval from ten to fifteen years is as follows:

1. The increase in risk on the total integrated plant risk as measured by person-rem/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 0.56% (0.004 person rem/yr). 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 large early release frequency (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 1.09×10^{-8}/yr. Since Regulatory Guide 1.174 defines very small changes in LERF as below 10^{-7}/yr, increasing the ILRT interval at FitzPatrick 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 27.4% to the overall plant risk. Alternatively stated, the contribution of sequences involving containment failure for the ten-year interval is 72.6%. These numbers are consistent with those documented in the FitzPatrick IPE. For the proposed fifteen-year interval, the contribution of sequences involving containment failure increased to 73.03%. Therefore, $\Delta CCFP_{10-15}$ is found to be 0.43%. This signifies a very small increase and represents a negligible change in the FitzPatrick containment defense-in-depth.

Additional risk considerations (external event hazards, potential containment liner corrosion) were also evaluated. These are summarized in Attachment 4.

Further assurance of containment leak-tight integrity is provided through periodic inservice inspections conducted in accordance with the requirements of the American Society of Mechanical Engineers Boiler and Pressure Vessel Code (ASME Code), Section XI. More specifically, Subsection IWE provides the rules and requirements for inservice inspection of Class MC pressure-retaining components and their integral attachments, and of metallic shell and penetration liners of Class CC pressure-retaining components and their integral attachments in light-water cooled plants. Furthermore, NRC regulations, 10 CFR 50.55a(b)(2)(ix)(E), require licensees to conduct visual inspections of the accessible areas of the containment 3 times every 10 years. These requirements are not affected by the extended ILRT interval. In addition, Appendix J, Type B local leak tests performed to verify the leak-tight integrity of containment penetration bellows, airlocks, seals, and gaskets are also not affected by the change to the Type A test frequency, as discussed more fully in the accompanying Safety Evaluation (Attachment 2).

Attachment 5 provides a description of the inservice inspection program for the FitzPatrick containment and a summary of recent inspection results. Inspections conducted to date have not identified any degradation or other condition that would threaten the structural integrity of the containment.

The NRC has approved similar risk-informed submittals relating to a one-time extension of a Type A test interval for a number of plants, including Carolina Power & Light Company's Brunswick Unit 1, Exelon Nuclear's Peach Bottom Unit 3, and Entergy's Indian Point 3 (IP3) nuclear power plant. The IP3 request was submitted on September 6, 2000 (IPN-00-062) and supplemented on January 18, 2001 (IPN-01-007) and on April 2, 2001 (IPN-01-030). The NRC approval was granted on April 17, 2001 (TAC No. MB0178).

The signed original of the Application for Amendment to the Operating License is enclosed for filing. Attachment 1 contains the proposed new TS pages and Attachment 2 is the Safety Evaluation for the proposed changes. A markup of the affected TS pages is included as Attachment 3. As previously indicated, Attachment 4 provides the supporting risk assessment evaluation, while Attachment 5 provides a synopsis of containment related inservice testing. There are no TS Bases associated with this request.

ENO requests approval of the proposed license amendment by August 10, 2004 with the amendment being implemented within thirty days following approval. The requested approval date and implementation period will allow sufficient time for effective planning and scheduling of affected activities associated with Refueling Outage 16, scheduled to begin on October 4, 2004.

In accordance with 10 CFR 50.91, a copy of this application, with attachments, is being provided to the designated New York State official.

These are no new commitments made in this letter. If you should have any questions regarding the submittal, please contact Mr. Andrew Halliday at (315) 349-6055.

I declare under penalty of perjury that the foregoing is true and correct.

Executed on this the 28 day of July 2003.

Very truly yours,

T. A. Sullivan
Site Vice President

Attachments:
1. Revised Technical Specification Pages
2. Safety Evaluation
3. Marked Up Technical Specification Pages
4. Plant Specific Risk Assessment Report
5. Containment Inservice Inspection Program Summary

cc: Regional Administrator, Region I
U. S. Nuclear Regulatory Commission
475 Allendale Road
King of Prussia, PA 19406

Office of the Resident Inspector
U. S. Nuclear Regulatory Commission
P. O. Box 136
Lycoming, NY 13093

Mr. G. Vissing, Project Manager
Project Directorate I
Division of Licensing Project Management
Office or Nuclear Reactor Regulation
U. S. Nuclear Regulatory Commission
Mail Stop: 8C2
Washington, DC 20555

Mr. Peter R. Smith, Acting President
New York State Energy, Research,
and Development Authority
Corporate Plaza West
286 Washington Avenue Extension
Albany, NY 12203-6399

ATTACHMENT 1 to JAFP-03-0108

REVISED TECHNICAL SPECIFICATION PAGES

Proposed License Amendment to Provide a One-time Integrated Leak Rate Test (ILRT) Interval Extension

Entergy Nuclear Operations, Inc.
JAMES A. FITZPATRICK NUCLEAR POWER PLANT
Docket No. 50-333
DPR-59

5.5 Programs and Manuals (continued)

5.5.6 Primary Containment Leakage Rate Testing Program

This program implements the leakage rate testing of the containment as required by 10 CFR 50.54(o) and 10 CFR 50. Appendix J. Option B. as modified by approved exemptions. This program shall be in accordance with the guidelines contained in Regulatory Guide 1.163. "Performance-Based Containment Leak-Test Program." dated September 1995. as modified by the following exceptions:

- NEI 94-01-1995. Section 9.2.3: The first Type A test performed after the March 7. 1995 Type A test shall be performed no later than March 7. 2010.
- Type C testing of valves not isolable from the containment free air space may be accomplished by pressurization in the reverse direction. provided that testing in this manner provides equivalent or more conservative results than testing in the accident direction. If potential atmospheric leakage paths (e.g.. valve stem packing) are not subjected to test pressure. the portions of the valve not exposed to test pressure shall be subjected to leakage rate measurement during regularly scheduled Type A testing. A list of these valves. the leakage rate measurement method. and the acceptance criteria. shall be contained in the Program.

a. The peak primary containment internal pressure for the design basis loss of coolant accident. P_a. is 45 psig.

b. The maximum allowable primary containment leakage rate. L_a. at P_a. shall be 1.5% of containment air weight per day.

c. The leakage rate acceptance criteria are:

1. Primary containment leakage rate acceptance criteria is $\leq 1.0\ L_a$. During plant startup following testing in accordance with this program. the leakage rate acceptance criteria are $\leq 0.60\ L_a$ for the Type B and Type C tests. and $\leq 0.75\ L_a$ for the Type A tests.

2. Air lock testing acceptance criteria are:

   (a) Overall air lock leakage rate is $\leq 0.05\ L_a$ when tested at $\geq P_a$: and

   (b) For each door seal. leakage rate is ≤ 120 scfd when tested at $\geq P_a$.

(continued)

ATTACHMENT 2 to JAFP-03-0108

SAFETY EVALUATION

Proposed License Amendment to Provide a One-time Integrated Leak Rate Test (ILRT) Interval Extension

Entergy Nuclear Operations, Inc.
JAMES A. FITZPATRICK NUCLEAR POWER PLANT
Docket No. 50-333
DPR-59

I. DESCRIPTION

This application for amendment to the James A. FitzPatrick (JAF) Technical Specifications (TS) proposes to revise Technical Specification 5.5.6 "Primary Containment Leakage Rate Testing Program". This revision takes a one time exception to the ten (10) year frequency of the performance-based leakage rate testing program for Type A tests as required by NEI 94-01 (Reference 1). The one time exception is to the requirement of NEI 94-01 to perform an integrated leak rate test (ILRT) at a frequency of up to ten years, with allowance for a fifteen month extension. The exception is to allow ILRT testing within fifteen years from the last ILRT, performed on March 7, 1995. This application represents a risk informed, cost beneficial licensing change. The integrated leak rate test imposes significant expense on the station while the safety benefit of performing it within ten years, versus fifteen years, is minimal. The specific change is as follows:

1. **TS Section 5.5.6, page 5.5-5**

   Replace:

   > "...as modified by the following exception:"

   With:

   > "...as modified by the following exceptions."

   Add:

   > "• NEI 94-01 - 1995, Section 9.2.3: The first Type A Test performed after the March 7, 1995 Type A test shall be performed no later than March 7, 2010."

   In addition, the existing paragraph beginning with "Type C testing of valves ... " is bulleted and indented for formatting consistency.

II. PURPOSE OF THE PROPOSED CHANGE

The current FitzPatrick ten year Type A test is due on March 7, 2005. This test is currently scheduled to be performed during refuel outage (RO), RO16, scheduled for October 2004. This one time exception will permit deferral of the test beyond RO16 to a later outage within the five year extension window. Deferring this test for an additional five (5) years will result in substantial cost savings associated with both direct costs for performing the test and indirect costs associated with critical path outage time. Cost savings have been conservatively estimated for this outage at $660,000, including a minimum reduction of thirty-six hours of critical path outage time.

III. SAFETY IMPLICATIONS OF THE PROPOSED CHANGE

1. BACKGROUND

10 CFR 50, Appendix J, Option B Requirements:

The testing requirements of 10 CFR 50, Appendix J, provide assurance that leakage through the containment, including systems and components that penetrate the containment, does not exceed the allowable leakage values specified in the Technical Specifications. The limitation of containment leakage provides assurance that the containment would perform its design function following an accident up to and including the plant design basis accident. Appendix J identifies three types of required tests: Type A tests, intended to measure the primary containment overall integrated leakage rate; Type B tests, intended to detect local leaks and to measure leakage across pressure-containing or leakage-limiting boundaries for primary containment penetrations; and Type C tests, intended to measure containment isolation valve leakage rates. Type B and C tests identify the vast majority of potential containment leakage paths. Type A tests identify overall (integrated) containment leakage rate and serve to ensure continued leakage integrity of the containment structure by evaluating those structural parts of the containment not covered by Type B and C testing.

10 CFR 50, Appendix J, was revised, effective October 26, 1995, to allow licensees to choose containment leakage testing under Option A "Prescriptive Requirements" or Option B "Performance-Based Requirements." In October 1996, Amendment 234 (Reference 2) was issued to the FitzPatrick Operating License to permit implementation of 10 CFR 50, Appendix J, Option B. Amendment 234 added a Technical Specification section (section 6.20 in custom Technical Specifications, now section 5.5.6 following conversion to standard Technical Specifications) requiring Type A, B and C testing in accordance with Regulatory Guide (RG) 1.163 (Reference 3). Regulatory Guide 1.163 specifies a method acceptable to the NRC for complying with Option B by approving the use of NEI 94-01 and ANSI/ANS 56.8 - 1994 (Reference 4), subject to several regulatory positions in the guide. NEI 94-01 specifies an initial Type A test interval of 48 months, but allows an extended interval of ten years, based upon two consecutive successful tests. There is also a provision for extending the test interval an additional fifteen months under certain circumstances.

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. Similarly, the proposed change to the Type A test frequency will not directly result in an increase in containment leakage.

The extended frequency interval for testing allowed by NEI 94-01 is based upon a generic evaluation documented in NUREG-1493, "Performance-Based Containment Leak-Test Program" (Reference 5). 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."

Exceptions to the requirements of RG 1.163, are allowed by 10 CFR 50, Appendix J, Option B, Section V.B, "Implementation," which states "The Regulatory Guide or other implementing document used by a licensee, or applicant for an operating license, to develop a performance based leakage-testing program must be included, by general reference, in the plant technical specifications. The submittal for technical specification revisions must contain justification, including supporting analyses, if the licensee chooses to deviate from methods approved by the Commission and endorsed in a regulatory guide." Since exceptions meeting the stated requirements are permitted, Technical Specification amendment applications satisfying these requirements do not require an exemption to Option B.

2. PLANT SPECIFIC INFORMATION

10 CFR 50 Appendix J, Option B Plant Specific Implementation

As previously stated, Amendment 234 to the FitzPatrick Operating License permitted implementation of 10 CFR 50, Appendix J, Option B for FitzPatrick. Amendment 234 requires Type A, B and C testing be conducted in accordance with Regulatory Guide (RG) 1.163, which in turn endorses the methodology for complying with Option B identified in NEI 94-01. The surveillance frequency for Type A testing in NEI 94-01 is at least once per ten years based on an acceptable performance history (i.e., two consecutive periodic Type A tests at least 24 months apart where the calculated performance leakage rate was less than $1.0L_a$) and consideration of the performance factors in NEI 94-01, Section 11.3. The two most recent Type A tests at JAF have been successful. The following extract from JAF-RPT-PC-02342, "James A. Fitzpatrick

Nuclear Power Plant Primary Containment Leakage Rate Testing Program Plan" (Reference 6), describes the test results:

"5.11 BASELINE CONTAINMENT EVALUATIONS

The performance leakage rates are calculated in accordance with NEI 94-01, Section 9.1.1. The performance leakage rate includes the Type A UCL plus the as-left minimum pathway leakage rate for all Type B and C pathways not in service, isolated, or not lined up in their test position. In addition, leakage pathways that were isolated during the performance of the test are included in the test results by adding the as-found minimum pathway leakage rate to the Type A UCL. The performance leakage rate does not include leakage savings (i.e., improvements to Type B and C components made prior to the Type A test).

For the June 1990 Type A test, the Total Time UCL leakage rate was 0.27045% wt./ day. The minimum pathway leakage rate for Type B and C pathways not in service and level corrections was 0.1331 % wt./day. The performance leakage rate was 0.27045 + 0.1331 = 0.4035 % wt./day, which was acceptable. There were no leakage pathways isolated during the performance of the test.

For the March 7, 1995 periodic Type A test, the total time UCL leakage rate, was 0.0394 % wt./day. The minimum pathway leakage rate for Type B and C pathways not in service was 0.023595 % wt./day. Therefore, the performance leakage rate was 0.0394 + 0.023595 = 0.062995 % wt./day."

These results compare with an acceptable design leakage rate for FitzPatrick (1.0 L_a) of 1.5 % wt./day. Based upon these two consecutive successful tests, the current ILRT interval requirement for JAF is ten years.

The results of the two previous ILRT tests conducted for FitzPatrick (May 1985 and April 1987) were also reviewed. Leakage rates for these tests were 0.281214 % wt./day and 0.304442 % wt./day respectively.

The results of these four tests conducted over an approximate ten year interval demonstrate consistent low leakage for the FitzPatrick primary containment structure.

Plant Testing and Inspection Programs

In addition to periodic Type A testing, various inspections and tests are routinely performed to assure primary containment integrity. These include Type B and C testing performed in accordance with Appendix J, Option B; inspection activities performed as part of the plant Inservice Inspection program; maintenance rule related inspections; and others. The aggregate results of these inspections serve to provide a high degree of assurance of continued primary containment integrity.

- **Type B and Type C Program**

The FitzPatrick Appendix J, Type B and Type C test programs are described in JAF-RPT-PC-02342, "James A. Fitzpatrick Nuclear Power Plant Primary Containment Leakage Rate Testing Program Plan." Regarding the scope of these programs, the program plan states, in part:

Electrical penetrations, airlocks, hatches, flanges, and valves within the scope of the Appendix J Program Plan and which are not exempt shall be tested in compliance with the requirements of 10 CFR 50 Appendix J Option B and Regulatory Guide 1.163.

The Type B and C test program provides a means to detect or measure leakage across pressure containing or leakage limiting barriers of the primary reactor containment. The results of the test program are used to ensure that proper maintenance and repairs are made on the primary reactor containment components over their service life. The Type B and C test program provides a means to protect the health and safety of plant personnel and the public by maintaining the leakage from these components below required levels.

The Type B and C test program consists of local leak rate testing of penetrations which utilize a resilient seal, expansion bellows, double gasketed manways, hatches, and flanges, drywell airlock, and containment isolation valves that serve as a barrier to the release of the post accident primary containment atmosphere. These components are tested with air or nitrogen at a pressure greater than or equal to 45 psig (P_a), not to exceed 56 psig (this does not account for instrument inaccuracies). Tests performed on-line will assure that full accident differential pressure is applied across the barrier under test, accounting for containment inerting, or system head pressure. The Main Steam Isolation Valves are tested at a greater than or equal to 25 psig per the technical specifications.

As previously noted, Type B and Type C testing evaluate all but a small portion of potential containment leakage pathways. Nothing in this amendment request affects the scope, performance or scheduling of Type B or Type C tests. These programs will continue to provide a high degree of assurance that primary containment integrity is maintained.

- **Inservice Inspection (ISI) Program**

Effective September, 1996, the NRC endorsed Subsections IWE and IWL of ASME Section XI, 1992 Edition including 1992 Addenda. These subsections contain inservice inspection and repair/replacement rules for Class MC and Class CC components. The reactor containment is a free-standing steel containment, to which the requirements of Subsection IWE apply. These requirements are included in the inservice inspection program for FitzPatrick, described in JAF-ISI-0002, "Third Inservice Inspection Interval Inservice Inspection Program" (Reference 7).

Attachment 5 provides a description of ISI program inspection activities for the FitzPatrick containment and a summary of recent inspection results. Basically, these inspections assess the condition of containment structural components and coatings, providing assurance against structural or material degradation. As such, these inspections complement Type A testing and provide a high degree of assurance of continued containment structural integrity.

- **Additional Tests and Inspections**

  Additional tests and inspections are conducted which assure the continued good material condition of the containment and associated containment integrity. These include Maintenance Rule inspections; periodic surveillance testing conducted, in addition to Type A, B, and C tests, as part of the plant Primary Containment Leakage Rate Testing Program; periodic walkdowns; and post maintenance tests. Attachment 5 provides further details.

Plant Operational Performance:

The James A. Fitzpatrick NPP is a power uprated 881 Mwe, General Electric Boiling Water Reactor (BWR4). The reactor is contained in a Chicago Bridge and Iron Works supplied Mark 1, Free Standing Steel Containment Building. The containment consists of two primary interconnected structures: the drywell, housing the reactor and related components, and a toroidal suppression chamber (torus). The drywell, which includes the major primary containment volume, is inerted with nitrogen and maintained at a nominal 1.7 psid positive pressure with respect to the torus. This pressure differential is required by Technical Specifications (LCO 3.2.6.4) and monitored by plant during instrumentation and through periodic surveillance (SR 3.6.2.1). The differential is initially established during drywell inerting by pressurizing the drywell using plant nitrogen. During plant operation, the combination of a small amount of normal instrument nitrogen leakage within the drywell and leak tightness of the containment structure is such that nitrogen typically does not have to be added to the drywell to maintain the required differential.

Plant Technical Specifications state:

3.6 CONTAINMENT SYSTEMS

3.6.2.4 Drywell-to-Suppression Chamber Differential Pressure

LCO 3.6.2.4   *The drywell pressure shall be maintained ≥ 1.7 psi above the pressure of the suppression chamber.*

------------------------NOTE------------------------
Not required to be met for 4 hours during Surveillances that cause or require the drywell-to-suppression chamber differential pressure to be outside the limit.

APPLICABILITY: MODE 1 during the time period:

a. *From 24 hours after THERMAL POWER is > 15% RTP following startup, to*

b. *24 hours prior to reducing THERMAL POWER to < 15% RTP prior to the next scheduled reactor shutdown.*

Required action if the differential pressure requirement is not met is to restore the parameter to within limits within eight hours or reduce reactor power to <15% within 12 hours. Surveillance testing requirements specify verifying the differential pressure within limit on a 12 hour surveillance frequency.

The major portion of the FitzPatrick containment is thus normally pressurized. Although the pressure is not as significant as that resulting from a Design Basis Accident, the fact that the containment is normally pressurized provides a degree of assurance of containment structural integrity (i.e. no large leak paths in the containment structure). Significant leakage would be identified using plant instrumentation or through increased nitrogen usage (periodically monitored) needed to maintain the required differential pressure, and would be investigated promptly and addressed within the scope of the plant Corrective Action system. This feature is a complement to periodic visual inspections of the interior and exterior of the containment structure, and serves to provide added assurance of structural integrity for those areas that may be inaccessible for visual examination.

3. PLANT SPECIFIC RISK ASSESSMENT

Attachment 4 contains a detailed, plant specific risk assessment performed in support of this amendment request. This assessment evaluates the risk impact of extending the Type A test interval for FitzPatrick from ten to fifteen years. The assessment complements the studies cited in NUREG-1493 that concluded that Type A testing intervals could be extended to as much as twenty years with negligible impact on risk.

The conclusions of the plant specific assessment are that effects on risk from the requested change are negligible or non-risk significant. Methodology and a summary of results are as follows:

- **Approach and Methodology:**

  In performing the risk assessment evaluation, 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 NRC Regulatory Guide 1.174, "An Approach for Using Probabilistic Risk Assessment In Risk-Informed Decisions On Plant-Specific Changes to the Licensing Basis" were used. 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 and the approach outlined in the Indian Point Unit Three Nuclear Power Plant ILRT extension submittal.

The risk assessment evaluation uses the current FitzPatrick Individual Plant Examination (IPE) internal events model that includes a Level 2[1] analysis of core damage scenarios and subsequent containment response resulting in various fission product release categories (including no release). The release category end states from the FitzPatrick Level 2 model are also applied to align with those used by the NRC in NUREG/CR-4551 for Peach Bottom Unit 2. This categorization allows the population dose information provided in NUREG/CR-4551 (adjusted by estimated changes in population since the publication of that document) to be used as a consequence model to provide an estimate of the person-rem dose per reactor year associated with various scenarios. The change in plant risk is then evaluated based on the potential change in population dose rate (person-rem/yr), change in Large Early Release Frequency (LERF), and the change in conditional containment failure probability (CCFP).

In addition to the internal events risk assessment evaluation, the impact associated with extending the Type A test frequency interval was further examined by considering external event hazard or potential containment liner corrosion. The purpose for these additional evaluations was 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.

- **Summary of Results:**

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

1) The increase in risk on the total integrated plant risk as measured by person-rem/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 0.56% (0.004 person rem/yr). 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 1.09×10^{-8}/ry. Since Regulatory Guide 1.174 defines very small changes in LERF as below 10^{-7}/yr, increasing the ILRT interval at FitzPatrick

[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.

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 27.4% to the overall plant risk. Alternatively stated, the contribution of sequences involving containment failure for the ten-year internal is 72.6%. These numbers are consistent with those documented in the FitzPatrick IPE. For the proposed fifteen-year interval, the contribution of sequences involving containment failure increased to 73.03%. Therefore, $\Delta CCFP_{10\text{-}15}$ is found to be 0.43%. This signifies a very small increase and represents a negligible change in the FitzPatrick containment defense-in-depth.

Additional risk considerations (external event hazards, potential containment liner corrosion) were also evaluated, with a similar conclusion that the requested test interval extension poses negligible risk. These evaluations are summarized in Attachment 4.

4. CONCLUSION

Previous Type A tests confirm that the JAF reactor containment structure exhibits extremely low leakage and represents minimal risk to increased leakage. The risk is minimized by continued Type B and Type C testing, reinforced by Inservice Inspection (ISI) program and Maintenance Rule inspections, by other periodic walkdowns and inspections, and by operating experience with a containment that normally operates at a positive pressure. These, in aggregate, provide continuing confidence in containment integrity.

This experience is supplemented by studies, including a plant specific risk analysis, that conclude that the risk associated with extending the Type A test interval on a one-time basis as requested is negligibly small.

It is therefore concluded that the cost-beneficial, risk informed change represented by this request is prudent and reasonable, and that the requested change involves no significant hazards as further documented in the following section.

IV. EVALUATION OF SIGNIFICANT HAZARDS CONSIDERATION

In accordance with the requirements of 10 CFR 50.92, the enclosed application is judged to involve no significant hazards based upon the following information:

1. **Does the change involve a significant increase in the probability or consequences of an accident previously analyzed?**

   The change <u>does not</u> involve a significant increase in the probability or consequences of an accident previously analyzed.

   The proposed revision to Technical Specifications adds a one time extension to the current interval for Type A testing. The current test interval of ten years, based on past performance, would be extended on a one time basis to fifteen years from the last Type A test. The proposed extension to Type A testing cannot increase the probability of an accident previously evaluated since the containment Type A testing extension is not a modification and the test extension is not of a type that could lead to equipment failure or accident initiation.

   The proposed extension to Type A testing does not involve a significant increase in the consequences of an accident since research documented in NUREG-1493 has found that, generically, very few potential containment leakage paths are not identified by Type B and C tests. The NUREG concluded that reducing the Type A (ILRT) testing frequency to one per twenty years was found to lead to an imperceptible increase in risk. These generic conclusions were confirmed by a plant specific risk analysis performed using the current FitzPatrick Individual Plant Examination (IPE) internal events model.

   Testing and inspection programs in place at FitzPatrick also provide a high degree of assurance that the containment will not degrade in a manner detectable only by Type A testing. The last four Type A tests show leakage to be below acceptance criteria, indicating a very leak tight containment. Type B and C testing required by Technical Specifications will identify any containment opening such as valves that would otherwise be detected by the Type A tests. Inspections, including those required by the ASME code and the maintenance rule are performed in order to identify indications of containment degradation that could affect that leak tightness.

   These factors in part and in aggregate show that a Type A test extension of up to five years will not represent a significant increase in the consequences of an accident.

2. **Does the change create the possibility of a new or different kind of accident from any accident previously analyzed?**

   The change <u>does not</u> create the possibility of a new or different kind of accident from any accident previously analyzed. The proposed revision to Technical Specifications adds a one time extension to the current interval for Type A testing. The current test interval of ten years, based on past performance, would be extended on a one time basis to fifteen years from the last Type A test. The proposed extension to Type A testing cannot create the possibility of a new or different type of accident since there are no physical changes being made to the plant and there are no changes to the operation of the plant that could introduce a new failure mode creating an accident or affecting the mitigation of an accident.

3. **Does the change involve a significant reduction in the margin of safety?**

   The change does not involve a significant reduction in the margin of safety. The proposed revision to Technical Specifications adds a one time extension to the current interval for Type A testing. The current test interval of ten years, based on past performance, would be extended on a one time basis to fifteen years from the last Type A test. The proposed extension to Type A testing will not significantly reduce the margin of safety. The NUREG 1493 generic study of the effects of extending containment leakage testing found that a 20 year extension in Type A leakage testing resulted in an imperceptible increase in risk to the public. NUREG -1493 found that, generically, the design containment leakage rate contributes about 0.1 percent to the individual risk and that the decrease in Type A testing frequency would have a minimal affect on this risk since 95% of the potential leakage paths are detected by Type C testing. This was further confirmed by a plant specific risk assessment using the current FitzPatrick Individual Plant Examination (IPE) internal events model that concluded the risk associated with this change is negligibly small and/or non-risk significant.

V. REFERENCES

1. NEI 94-01, "Industry Guideline for Implementing Performance-Based Option of 10 CFR Part 50, Appendix J"
2. Amendment No. 234 to Facility Operating License No. DPR-59 (TAC No. M95099)
3. Regulatory Guide (RG) 1.163, "Performance-Based Containment Leak-Test Program"
4. ANSI/ANS 56.8 – 1994, "Containment System Leakage Testing Requirements"
5. NUREG-1493, "Performance-Based Containment Leak-Test Program"
6. JAF-RPT-PC-02342, "James A. Fitzpatrick Nuclear Power Plant Primary Containment Leakage Rate Testing Program Plan"
7. JAF-ISI-0002, "Third Inservice Inspection Interval Inservice Inspection Program"

ATTACHMENT 3 to JAFP-03-0108

MARKED UP TECHNICAL SPECIFICATION PAGES

Proposed License Amendment to Provide a One-time Integrated Leak Rate Test (ILRT) Interval Extension

Entergy Nuclear Operations, Inc.
JAMES A. FITZPATRICK NUCLEAR POWER PLANT
Docket No. 50-333
DPR-59

5.5 Programs and Manuals (continued)

5.5.6 Primary Containment Leakage Rate Testing Program

Insert A

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

exceptions:

Bullet and indent (format change only)

Type C testing of valves not isolable from the containment free air space may be accomplished by pressurization in the reverse direction, provided that testing in this manner provides equivalent or more conservative results than testing in the accident direction. If potential atmospheric leakage paths (e.g., valve stem packing) are not subjected to test pressure, the portions of the valve not exposed to test pressure shall be subjected to leakage rate measurement during regularly scheduled Type A testing. A list of these valves, the leakage rate measurement method, and the acceptance criteria, shall be contained in the Program.

a. The peak primary containment internal pressure for the design basis loss of coolant accident, P_a, is 45 psig.

b. The maximum allowable primary containment leakage rate, L_a, at P_a, shall be 1.5% of containment air weight per day.

c. The leakage rate acceptance criteria are:

 1. Primary containment leakage rate acceptance criteria is $\leq 1.0\ L_a$. During plant startup following testing in accordance with this program, the leakage rate acceptance criteria are $\leq 0.60\ L_a$ for the Type B and Type C tests, and $\leq 0.75\ L_a$ for the Type A tests.

 2. Air lock testing acceptance criteria are:

  (a) Overall air lock leakage rate is $\leq 0.05\ L_a$ when tested at $\geq P_a$; and

  (b) For each door seal, leakage rate is ≤ 120 scfd when tested at $\geq P_a$.

(continued)

Insert A

- NEI 94-01-1995. Section 9.2.3: The first Type A test performed after the March 7. 1995 Type A test shall be performed no later than March 7. 2010.

ATTACHMENT 4 to JAFP-03-0108

PLANT SPECIFIC RISK ASSESSMENT REPORT

Proposed License Amendment to Provide a One-time Integrated Leak Rate Test (ILRT) Interval Extension

Entergy Nuclear Operations, Inc.
JAMES A. FITZPATRICK NUCLEAR POWER PLANT
Docket No. 50-333
DPR-59

Entergy

ENTERGY NUCLEAR NORTHEAST
Engineering Report Cover Sheet

Engineering Report Title:

Risk Impact Assessment of Extending Containment Type A Test Interval

Engineering Report Type:

New ☒ Revision ☐ Cancelled ☐ Superceded ☐

Applicable Site(s)

IP1 ☐ IP2 ☐ IP3 ☐ JAF ☒ PNPS ☐ VY ☐

Quality-Related: ☒ Yes ☐ No

Prepared by: John Favara / Kou-John Hong (Responsible Engineer) Date: 6-10-03

Verified/ Reviewed by: John Bretti (Design Verifier/Reviewer) Date: 6-10-03

Approved by: Clem Yeh (Supervisor) Date: 6-11-03

Multiple Site Review

Site	Design Verifier/Reviewer	Supervisor	Date

RECORD OF REVISIONS

Revision No.	Description of Change	Reason For Change
0	Original report	NA

TABLE OF CONTENTS

TABLE OF CONTENTS (continued)

LIST OF FIGURES

LIST OF TABLES

LIST OF TABLES

EXECUTIVE SUMMARY

Revisions to 10CFR 50, Appendix J allow individual plants to extend Type A surveillance testing requirements 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 was less than normal containment leakage or $1.0L_a$. James A. FitzPatrick (FitzPatrick) selected the revised requirements as its testing program. FitzPatrick's current ten-year Type A test frequency is due to be performed during the upcoming refueling outage 16 (RF16), scheduled for October 4, 2004. Prior to the performance of that test, however, FitzPatrick is seeking an extension of the test interval to fifteen years. A substantial cost savings will be realized and unnecessary personnel radiation exposure will be avoided by deferring the Type A test for an additional five years. Cost savings have been estimated for this outage at approximately $660,000.00, which includes labor, equipment and critical path outage time needed to perform the test. 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 RO-16.

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 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 NRC Regulatory Guide 1.174, "An Approach for Using Probabilistic Risk Assessment In Risk-Informed Decisions On Plant-Specific Changes to the Licensing Basis" were used. 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 and the approach outlined in the Indian Point Unit Three Nuclear Power Plant ILRT extension submittal.

The risk assessment evaluation uses the current FitzPatrick Individual Plant Examination (IPE) internal events model that includes a Level 2[1] analysis of core damage scenarios and subsequent containment response resulting in various fission product release categories (including no release). The release category end states from the FitzPatrick Level 2 model are also applied to align with those used by the NRC in NUREG/CR-4551 for Peach Bottom Unit 2. This categorization allows the population dose information provided in NUREG/CR-4551 (adjusted by estimated changes in population since the publication of that document) to be used as a consequence model to provide an estimate of the person-rem dose per reactor year associated with various scenarios. The change in plant risk is then evaluated based on the potential change in population dose rate (person-rem/ry), change in Large Early Release Frequency (LERF), and the change in conditional containment failure probability (CCFP).

The risk assessment evaluation examined FitzPatrick's IPE 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).

[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.

- 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 conclusion of the plant internal events risk associated with extending the Type A ILRT interval from ten to fifteen years is 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 0.56% (0.004 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 1.09 x 10^{-8}/ry. Since Regulatory Guide 1.174 defines very small changes in LERF as below 10^{-7}/yr, increasing the ILRT interval at FitzPatrick 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 27.4% to the overall plant risk. Alternatively stated, the contribution of sequences involving containment failure for the ten-year interval is 72.6%. These numbers are consisted with those documented in the FitzPatrick IPE. For the proposed fifteen-year interval, the contribution of sequences involving containment failure increased to 73.03%. Therefore, $\Delta CCFP_{10-15}$ is found to be 0.43%. This signifies a very small increase and represents a negligible change in the FitzPatrick 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 FitzPatrick ILRT frequency from 1-in-10 years to 1-in-15 years is 1.03 x 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 [6]. Therefore, increasing the ILRT interval at FitzPatrick 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 0.16% (0.038 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 0.20%. A change in ΔCCFP 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 FitzPatrick 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 [24] 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 1.22×10^{-8}/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 4.1×10^{-3} person-rem/ry or 0.57% 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 0.6%.

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 (2)	2.91×10^{3}	4.49×10^{-7}	1.31×10^{-3}	3.29×10^{-7}	9.58×10^{-4}
2	Containment Isolation System Failure	6.79×10^{5}	2.44×10^{-9}	1.66×10^{-3}	2.44×10^{-9}	1.66×10^{-3}
3a	Small Pre-Existing Failures (2), (3)	2.91×10^{4}	2.20×10^{-7}	6.40×10^{-3}	3.29×10^{-7}	9.60×10^{-3}
3b	Large Pre-Existing Failures (2), (3)	1.02×10^{5}	2.20×10^{-8}	2.24×10^{-3}	3.29×10^{-8}	3.36×10^{-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)(4)	6.11×10^{5}	1.74×10^{-7}	1.06×10^{-1}	1.74×10^{-7}	1.06×10^{-1}
7b	Containment Failure Due to Severe Accident (b) (4)	3.96×10^{5}	7.37×10^{-7}	2.91×10^{-1}	7.37×10^{-7}	2.91×10^{-1}
7c	Containment Failure Due to Severe Accident (c) (4)	5.21×10^{5}	5.82×10^{-8}	3.04×10^{-2}	5.82×10^{-8}	3.04×10^{-2}
7d	Containment Failure Due to Severe Accident (d) (4)	3.06×10^{5}	4.83×10^{-7}	1.48×10^{-1}	4.83×10^{-7}	1.48×10^{-1}
8	Containment Bypass Accidents	4.46×10^{5}	2.97×10^{-7}	1.32×10^{-1}	2.97×10^{-7}	1.32×10^{-1}
TOTALS:			2.44×10^{-6}	0.720	2.44×10^{-6}	0.724
Increase in Dose Rate						0.56%
Increase in LERF					1.09×10^{-8}	
Increase in CCFP (%)					0.43%	

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 [(2)]	2.91×10^{3}	2.69×10^{-6}	7.85×10^{-3}	1.56×10^{-6}	4.55×10^{-3}
2	Containment Isolation System Failure	6.79×10^{5}	5.19×10^{-8}	3.53×10^{-2}	5.19×10^{-8}	3.53×10^{-2}
3a	Small Pre-Existing Failures [(2), (3)]	2.91×10^{4}	2.06×10^{-6}	6.00×10^{-2}	3.09×10^{-6}	9.00×10^{-2}
3b	Large Pre-Existing Failures [(2), (3)]	1.02×10^{5}	2.06×10^{-7}	2.10×10^{-2}	3.09×10^{-7}	3.15×10^{-2}
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)[(4)]	6.11×10^{5}	2.45×10^{-5}	1.50×10^{1}	2.45×10^{-5}	1.50×10^{1}
7b	Containment Failure Due to Severe Accident (b) [(4)]	3.96×10^{5}	1.57×10^{-5}	6.20	1.57×10^{-5}	6.20
7c	Containment Failure Due to Severe Accident (c) [(4)]	5.21×10^{5}	1.24×10^{-6}	6.46×10^{-1}	1.24×10^{-6}	6.46×10^{-1}
7d	Containment Failure Due to Severe Accident (d) [(4)]	3.06×10^{5}	1.82×10^{-6}	5.58×10^{-1}	1.82×10^{-6}	5.58×10^{-1}
8	Containment Bypass Accidents	4.46×10^{5}	3.65×10^{-6}	1.63	3.65×10^{-6}	1.63
TOTALS:			5.19×10^{-5}	24.13	5.19×10^{-5}	24.17
Increase in Dose Rate						0.16%
Increase in LERF					1.03×10^{-7}	
Increase in CCFP (%)					0.198%	

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 (2)	2.91×10^{3}	4.49×10^{-7}	1.31×10^{-3}	3.26×10^{-7}	9.52×10^{-4}
2	Containment Isolation System Failure	6.79×10^{5}	2.44×10^{-9}	1.66×10^{-3}	2.44×10^{-9}	1.66×10^{-3}
3a	Small Pre-Existing Failures (2), (3)	2.91×10^{4}	2.20×10^{-7}	6.40×10^{-3}	3.29×10^{-7}	9.60×10^{-3}
3b	Large Pre-Existing Failures (2), (3)	1.02×10^{5}	2.29×10^{-8}	2.34×10^{-3}	3.51×10^{-8}	3.58×10^{-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)(4)	6.11×10^{5}	1.74×10^{-7}	1.06×10^{-1}	1.74×10^{-7}	1.06×10^{-1}
7b	Containment Failure Due to Severe Accident (b) (4)	3.96×10^{5}	7.37×10^{-7}	2.91×10^{-1}	7.37×10^{-7}	2.91×10^{-1}
7c	Containment Failure Due to Severe Accident (c) (4)	5.21×10^{5}	5.82×10^{-8}	3.04×10^{-2}	5.82×10^{-8}	3.04×10^{-2}
7d	Containment Failure Due to Severe Accident (d) (4)	3.06×10^{5}	4.83×10^{-7}	1.48×10^{-1}	4.83×10^{-7}	1.48×10^{-1}
8	Containment Bypass Accidents	4.46×10^{5}	2.97×10^{-7}	1.32×10^{-1}	2.97×10^{-7}	1.32×10^{-1}
TOTALS:			2.44×10^{-6}	0.720	2.44×10^{-6}	0.724
Increase in Dose Rate						0.57%
Increase in LERF					1.22×10^{-8}	
Increase in CCFP (%)					0.60%	

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

1) The population dose associated with the Technical Specification Leakage is based on scaling the population data, the power level, and allowable Technical Specification leakage compared to the Peach Bottom Unit NUREG/CR-4551 reference plant.

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

3) 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.

4) EPRI Class 7, containment failure due to severe accident, was subdivided into four subgroups based on FitzPatrick 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
CDFM	Conservative Deterministic Failure Margin
CET	Containment Event Tree
CF	Containment Failure
DCH	Direct Containment Heating
DW	Drywell
EPRI	Electrical Power Research Institute
EVNTRE	Event Progress Analysis Code
HCLPF	High Confidence Low Probability of Failure
ILRT	Integrated Leak Rate Testing
IPE	Individual Plant Examination
IPEEE	Individual Plant Examination for External Events
ISLOCA	Interface System Loss of Coolant Accident
IP3	Indian Point Unit Three Nuclear Power Plant
JAF	James A. FitzPatrick Nuclear Power Plant
LERF	Large Early Release Frequency
LLRT	Local Leak Rate Testing
LOCA	Loss of Coolant Accident
NEI	Nuclear Energy Institute

Nomenclature (continued)

NRC	United States Nuclear Regulatory Commission
MFCR	Mean Factional Contribution to Risk
PDS	Plant Damage State
PRA	Probabilistic Risk Analysis
PSA	Probabilistic Safety Assessment
RAI	Request for Additional Information
RCS	Reactor Coolant System
RPV	Reactor Pressure Vessel
RF	Refueling Outage
SCDF	Seismic Core Damage Frequency
SMA	Seismic Margin Assessment
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.

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.

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.

Definitions (continued)

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 James A. FitzPatrick (FitzPatrick) Technical Specification 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" [2], NEI's "Interim Guidance for Performing Risk Impact Assessments In Support of One-Time Extensions for Containment Integrated Leakage Rate Test Surveillance Intervals" [3], NEI 's "Additional Information for ILRT Extensions" [4], and the NRC regulatory guidance on the use of Probabilistic Risk Assessment (PRA) findings and risk insights in support of a request for a change in a plant's licensing basis as outlined in Regulatory Guide 1.174, "An Approach for Using Probabilistic Risk Assessment In Risk-Informed Decisions On Plant-Specific Changes to the Licensing Basis" [5].

In addition, the results and findings from the FitzPatrick Individual Plant Examination (IPE) update [6] are used for this risk assessment report.

1.2 Background

In October 26, 1995, the Nuclear Regulatory Commission (NRC) revised 10 CFR 50, Appendix J. The revision to Appendix J provided a performance based option, Option B "Performance-Based Requirements", for leakage-rate testing of light-water-cooled containments.

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, FitzPatrick selected the requirements under Option B as its testing program. FitzPatrick's current ten-year Type A test is due to be performed during refueling outage sixteen (RF16, scheduled for October 4, 2004). However, FitzPatrick seeks a one-time exemption based on the substantial cost savings of $660,000.00 from extending the test from the RF16 schedule to RF18. 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 RO-16.

The basis for the current 10-year test interval is provided in Section 11.0 of NEI 94-01, Revision 0, which was established in 1995 during development of the performance-based Option B to Appendix J [1]. Section 11.0 of NEI 94-01 states that NUREG-1493, "Performance-Based Containment Leak-Test Program," [7] dated September 1995, provides the technical basis to support rulemaking to revise leakage rate testing requirements contained in Option B to Appendix J.

The NUREG-1493 [7] report examined the impact of containment leakage on public health and safety associated with a range of extended leakage rate test intervals. The NUREG 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 FitzPatrick's), 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 [2]. The EPRI Methodology [2] 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 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.

Building upon the methodology of the EPRI TR-104285 [2] 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 [9] considered this an improvement on the EPRI study. Similar information is contained in the Crystal River Nuclear Power Plant submittal [10].

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" [3], and "Additional Information for ILRT Extensions," [4] that builds on the EPRI TR-104285 [2], IP3 [8] and Crystal River submittal [10] methodology and is intended to provide for more consistent submittals to the NRC.

The FitzPatrick 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 [5] 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 [5] 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 [5] 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 FitzPatrick risk assessment analysis uses the approach outlined in the Indian Point Unit Three Nuclear Power Plant (IP3) methodology [8], EPRI's TR-104285 [2], NEI's Interim Guidance [3], NEI 's "Additional Information for ILRT Extensions" [4], and the NRC regulatory guidance on the use of PRA findings and risk insights in support of a request for a change in a plant's licensing basis as outlined in Regulatory Guide 1.174 [5].

The EPRI TR-104285 methodology [2] involves a quantitative evaluation on the change in public risk of the affect of extending the ILRT and Local Leak Rate Test (LLRT) intervals. The EPRI TR-104285 study combined IPE Level 2[2] models with NUREG-1150 "Severe Accident Risks: An Assessment for Five U.S. Nuclear Power Plants" [11] Level 3[3] population dose models to perform the analysis. This study also used the approach of NUREG-1493 [7] in calculating the increase in pre-existing leakage probability due to extending the ILRT and LLRT test intervals.

EPRI TR-104285 used a simplified containment event tree (CET) to provide a risk-based framework for evaluating the effects of containment isolation failures impacted by Appendix J penetrations testing requirements. The CET regrouped core damage accident sequences into eight accident classes of containment response. These eight accident classes are:

1) Containment intact and isolated
2) Containment isolation failures due to support system or active failures
3) Type A (ILRT) related containment isolation failures
4) Type B (LLRT) related containment isolation failures
5) Type C (LLRT) related containment isolation failures
6) Other penetration related containment isolation failures
7) Containment failure due to core damage accident phenomena
8) Containment bypass

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.

The IP3 methodology [8] modified the EPRI TR-104285 [2] accident class 3 to include the probability of a containment breach (due to excessive leakage) at the time of core damage. Class 3 was redefined as accident sequence endstates 3a (small containment breach) and 3b (large containment breach). This reclassification resulted in the change in risk associated with changes in ILRT intervals (since a pre-existing leak is assumed to be present for these endstates). Furthermore, by changing the probability of a pre-existing leak, an evaluation of the change in large early release frequency (LERF) can be performed. The NRC, "Indian Point Nuclear Generating Station Unit No. 3 – Issuance of Amendment Re: Frequency of Performance-Based Leakage Rate Testing, April 17, 2001" [9] considered this an

---

[2] 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.

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

improvement on the EPRI study. Similar information is contained in the Crystal River submittal, "Supplemental Risk Informed Information in Support of License Amendment Request No. 267" [10].

NEI's Interim Guidance documents [3 and 4] improve on the above methods. Therefore, the FitzPatrick risk assessment analysis uses the approach outlined in the NEI's Interim Guidance [3] and NEI 's "Additional Information for ILRT Extensions" [4]. The nine steps of the methodology are:

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 latest FitzPatrick IPE Level 1[4] and Level 2 [6] 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. In order to assess the impact on offsite dose, Peach Bottom information, "Evaluation of Severe Accident Risks: Peach Bottom, Unit 2" [12] was used to estimate the FitzPatrick offsite dose.

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, 9] and other subsequent extensions [10].

The eighth step in the interim methodology calculates the change in LERF and compares it to the guidelines in Regulatory Guide 1.174 [5]. 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.

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

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 FitzPatrick is once per ten years [13].

2) The FitzPatrick (Revision 1) Level 1 and Level 2 internal events IPE models provide representative results for the analysis [6].

3) Radionuclide release categories defined in this report are consistent with the EPRI TR-104285 methodology. [2]

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 impact by the change in ILRT frequency.

6) The maximum containment leakage for Class 1 sequences is 1La [2]. (La is the Technical Specification maximum allowable containment leakage rate).

7) The maximum containment leakage for Class 3a sequences per the NEI Interim Guidance [3] and previously approved methodology [8, 9] is 10La.

8) The maximum containment leakage for Class 3b sequences per the NEI Interim Guidance [3] and previously approved methodology [8, 9] is 35La.

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

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 [3] and previously approved methodology [2, 8, and 9]. Furthermore, these releases have a break opening of greater than 0.5-inch but less than 2-inch diameter [8, 9].

11) Containment leak rates greater than 35La indicates a containment breach. This leak rate is considered 'large' per the NEI Interim Guidance [3] and previously approved methodology [8, 9].

12) Containment leak rates less than 2La indicates an intact containment. This leak rate is considered as 'negligible' per the NEI Interim Guidance [3] and previously approved methodology [8, 9].

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 [2] as augmented by NEI Interim Guidance [3, 4].

16) Although the meteorology data could play a role in the early health effects calculations, the meteorology and site topography for Peach Bottom Unit 2 and FitzPatrick are assumed to be sufficiently similar that any differences are assumed not to play a significant role in this evaluation of total population dose.

2.3 Data and Design Criteria

1) The FitzPatrick Level 1 and 2 IPE update is used as input to this analysis reflects the as built, as-operated plant. [6]

2) The CDF value, as reported in the Fitzpatrick IPE, Revision 1 is 2.44 x 10^{-6}/ry. [6]

3) The FitzPatrick Level 2 IPE is used to calculate the release frequencies for the accidents evaluated in this assessment[5]. Table 2-1 summarizes the FitzPatrick Level 1 IPE internal mean frequency results by core damage plant damage states (PDS). (See Attachment A)

4) Table 2-2 summarizes the pertinent FitzPatrick Level 2 IPE results in terms of release modes. The total release frequency is 1.75 x 10^{-6}/ry; with a total CDF of 2.44 x 10^{-6}/ry. Table 2-3 summarizes the correlation of the FitzPatrick Level 2 IPE results for containment failure accident progression bins in terms of release magnitudes. (See Attachment A)

5) The random large containment isolation failure probability, from the FitzPatrick IPE, Revision 1, Section 4.5 [6] is = 10^{-3}. This value reflects the mean probability of a solenoid valve failing to close on demand. This value is bounding, since other pathways involve redundant closed pathways with failure probabilities less than 10^{-6}.

6) 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 [3]. From the IP3 submittal, the probability that a liner leak will be small made use of the data presented in NUREG-1493 [7]. 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 [3] 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.

7) The conditional failure probability of having a large pre-existing containment leak is 0.0027. This value is derived from the NEI Interim Guidance [3]. It's based on the Jeffreys non-informative prior distribution[6] for zero failures. The formula is as follows:

---

[5] The Level 2 analysis used a point estimate CDF of 2.2 x 10^{-6}/ry. However, this analysis uses the mean CDF value in calculating the eight accident classes' frequencies.

[6] 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.

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

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

8) The total population dose risk for Peach Bottom Unit 2 is taken from NUREG/CR-4551, Table 5.1-1 [12]. This value is 7.9 person-rem per reactor year.

9) The mean internal event core damage frequency for Peach Bottom Unit 2 is taken from NUREG/CR-4551, Figure S-3 [12]. This value is 4.34×10^{-6} per reactor year.

10) The population dose data in NUREG/CR-4551 for Peach Bottom Unit 2 [12] is reported in ten distinct collapsed accident progression bins (APBs). These collapsed APBs are composed essentially of five characteristics: the occurrence of core damage,[7] the occurrence of vessel breach, primary system pressure at vessel breach, the location of containment failure, and the timing of containment failure. A description of these characteristics and their associated attributes are presented in Table 2-4.

11) The internal events mean fractional contribution to risk (MFCR) for each collapsed accident progression bin (APB) for Peach Bottom Unit 2 is taken from NUREG/CR-4551, Table 5.2-3 [12]. These are as follows:

Collapsed APB	MFCR
1	0.0210
2	0.0066
3	0.5560
4	0.2260
5	0.0022
6	0.0590
7	0.1180
8	0.0005
9	0.0100
10	0

12) The internal events individual conditional probabilities for each collapsed APB for Peach Bottom Unit 2 is taken from NUREG/CR-4551, Figure S-3 [12]. These are as follows:

Collapsed APB	Individual Conditional Probabilities
1	0.0220
2	0.0110
3	0.3410
4	0.1830
5	0.0030
6	0.0470
7	0.1100
8	0.1840
9	0.0890
10	0.0100

[7] Core damage here implies substantial core melt and relocation inside the reactor pressure vessel.

13) The 50-mile radius Peach Bottom Unit 2 population data used to characterize the population dose calculations is 3.02×10^6 [14]. This value is based on the population data presented in the Peach Bottom Unit 2 reactor risk study presented below.

NUREG/CR-4551 Peach Bottom Unit 2 Population Data at Different Radii From the Plant [12]

Distance From Plant		Population
(Km)	(Miles)	
1.6	1.0	118
4.8	3.0	1822
16.1	10.0	28,647
48.3	30.0	989,356
160.9	100.0	14,849,112
563.3	350.0	68,008,584
1609.3	1000.0	154,828,144

The analysis utilizes an estimate of the population density within 50 miles for the Peach Bottom Unit 2 plant. (See Appendix C of Reference 14 for more details). The value of 3.02×10^6 is used to relate that 50-mile population dose calculation from Peach Bottom Unit 2 to FitzPatrick; the population information is needed to appropriately scale the calculated dose from Peach Bottom Unit 2 to FitzPatrick.

14) The 50-mile radius FitzPatrick population data is obtained from a study performed for the Nine Mile Point and James A. FitzPatrick Nuclear Stations "Ingestion Pathway Population" [15]. The value used to characterize the population dose calculations is 8.98×10^5 [15]. This value is based on the information contain in Table M-2 [15] and reproduced below.

Nine Mile Point and James A. FitzPatrick Nuclear Stations Population Estimate [15]

ZONE	MILES FROM FACILITY	POPULATION 2000	2009	2010
1	0-1	8	8	8
2	1–2	342	343	344
3	2–3	926	930	931
4	3–4	1,454	1,460	1,461
5	4–5	1,422	1,428	1,429
6	5–6	4,326	4,343	4,348
7	6–7	9,251	9,288	9,297
8	7–8	10,795	10,834	10,849
9	8–9	5,155	5,176	5,181
10	9–10	4,553	4,571	4,576
15	10–15	32,849	32,980	33,012
20	15–20	28,433	28,546	28,574
25	20–25	45,046	44,285	44,097
30	25–30	109,672	107,819	107,361
35	30–35	166,335	162,885	162,034
40	35–40	219,025	214,482	213,361
45	40–45	131,252	130,561	130,389
50	45–50	127,023	126,354	126,188
	TOTAL:	897,867	886,293	883,440
POPULATION WITHIN 5 MILES:		4,152	4,169	4,173
POPULATION WITHIN 10 MILES:		38,232	38,381	38,424
POPULATION WITHIN 25 MILES:		144,560	142,118	144,107

15) The Peach Bottom Unit 2 reactor power level used in the NUREG/CR-4551 [12] consequence calculations is 3293 MWth.

16) The Peach Bottom Unit 2 maximum allowed containment leakage used in the NUREG/CR-4551 consequence calculations is 0.5 volume%/day [12].

17) The Peach Bottom Unit 2 containment volume used in the NUREG/CR-4551 [12] consequence calculations is 307,000 ft^3.

18) The FitzPatrick reactor power level used in this report is the power uprated adjusted value of 2536MWth [16].

19) The FitzPatrick technical specification maximum allowed containment leakage is 1.5 volume%/day [17].

20) The Fitzpatrick containment free volume used in this report is 264,000 ft^3 (drywell free volume of approximately 150,000 ft^3 and torus free volume of approximately 114,000 ft^3) [18].

2.4 Internal Events Impact

This section provides a step-by-step summary of the NEI guidance [3] as applied to the James A. FitzPatrick ILRT interval extension risk assessment. Each subsection addresses a step in the NEI guideline [3].

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 [2].

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 [3], 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:

CLASS_1_FREQUENCY = NCF - CLASS_3a_FREQUENCY - 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
= 6.91×10^{-7} /ry [Table 2-2]

CLASS_3a_FREQUENCY = frequency of small pre-existing containment liner leakage
= 6.59×10^{-8}/ry [See below write-up]

CLASS_3b_FREQUENCY = frequency of large pre-existing containment liner leakage
= 6.59×10^{-9}/ry [See below write-up]

Therefore:

$$\text{CLASS\_1\_FREQUENCY} = 6.91 \times 10^{-7} - 6.59 \times 10^{-8} - 6.59 \times 10^{-9}$$

$$\text{CLASS\_1\_FREQUENCY} = 6.19 \times 10^{-7}/\text{ry}$$

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)
= 10^{-3} [Section 2.3, input#5]

CDF = FitzPatrick PE core damage frequency = 2.44×10^{-6}/ry [Section 2.3, input #2]

Therefore:

$$CLASS\_2\_FREQUENCY = 10^{-3} * 2.44 \times 10^{-6}$$

$$CLASS\_2\_FREQUENCY = 2.44 \times 10^{-9}/ry$$

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 [3], the frequency per reactor year for this category is calculated as:

$$CLASS\_3a\_FREQUENCY = PROB_{class\_3a} * CDF$$

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#6]

CDF = FitzPatrick PE core damage frequency = 2.44×10^{-6}/ry [Section 2.3, input#2]

Therefore,

$$CLASS\_3a\_FREQUENCY = 0.027 * 2.44 \times 10^{-6}$$

$$CLASS\_3a\_FREQUENCY = 6.59 \times 10^{-8}/ry$$

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 [3], the frequency per reactor year for this category is calculated as:

$$CLASS\_3b\_FREQUENCY = PROB_{class\_3b} \times CDF$$

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 #7]

CDF = FitzPatrick PE core damage frequency = 2.44×10^{-6} ry [Section 2.3, input # 2]

Therefore,

$$CLASS\_3a\_FREQUENCY = 0.0027 * 2.44 \times 10^{-6}$$

$$CLASS\_3a\_FREQUENCY = 6.59 \times 10^{-9}/ry$$

<u>Frequency of EPRI Class 4 Sequences.</u> 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 [3], 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.

<u>Frequency of EPRI Class 5 Sequences.</u> 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 [3], because these failures are detected by Type C tests, this group is not evaluated any further.

<u>Frequency of EPRI Class 6 Sequences.</u> 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 [3], because these failures are detected by Type B or C tests, this group is not evaluated any further.

<u>Frequency of EPRI Class 7 Sequences.</u> 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 [3], the frequency per reactor year for this class is based on the plant Level 2 PSA results.

Because the Fitzpatrick IPE Level 2 containment failure results are summarized into four different release bins (Table 2-3), EPRI Class 7 is sub-divided in this report to reflect this sub-division of the FitzPatrick Level IPE 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 FitzPatrick Level 2 IPE early drywell failures release bin. Based on the FitzPatrick Level 2 IPE results summarized earlier in Table 2-3, the frequency of Category 7a is 1.74×10^{-7}/ry.

The frequency of Category 7b is the total frequency of the FitzPatrick Level 2 IPE early torus failures release bin. Based on the FitzPatrick Level 2 IPE results summarized earlier in Table 2-3, the frequency of Category 7b is 7.37×10^{-7}/ry.

The frequency of Category 7c is the total frequency of the FitzPatrick Level 2 IPE late drywell failures release bin. Based on the FitzPatrick Level 2 IPE results summarized earlier in Table 2-3, the frequency of Category 7c is 5.82 x 10^{-8}/ry.

The frequency of Category 7d is the total frequency of the FitzPatrick Level 2 IPE late torus failures release bin. Based on the FitzPatrick Level 2 IPE results summarized earlier in Table 2-3, the frequency of Category 7d is 4.83 x 10^{-7}/ry.

Frequency of EPRI Class 8 Sequences. 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 Fitzpatrick Level 1 IPE results summarized earlier in Table 2-3, the frequency of Class 8 is 2.97 x 10^{-7}/ry.

Note: for this class 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 [3]. 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 [3] 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 FitzPatrick La is1.5% of containment air weight per day (per FitzPatrick Technical Specification).

By definition, and per the NEI Interim Guidance [3] 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 2 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 [3] recommends two options for calculating population dose for the EPRI accident classes:

- Use of NUREG-1150 dose calculations [11]
- Use of plant-specific dose calculations

The use of generic dose information for NUREG-1150 [11] is recommended by NEI to make the ILRT risk assessment methodology more usable for plants that do not have a Level 3 PRA. Because FitzPatrick does not have a Level 3 PRA or associated plant-specific dose calculations for the EPRI's accident classes, this calculation uses NUREG-1150 dose results. Specifically, the doses for Peach Bottom Unit 2, as documented in NUREG/CR-4551 [12] are used.

The following substeps describes the methodology use for obtaining population dose result estimates for FitzPatrick using information from NUREG/CR-4551 for Peach Bottom Unit 2 [12]:

- Calculate Peach Bottom Unit 2 NUREG/CR-4551 Study Population Dose
- Calculate FitzPatrick Person-Rem Per Collapsed Accident Progression Bin (APB)
- Calculate FitzPatrick Person-Rem Frequency by Accident Progression Bin (APB)
- Calculate EPRI 's TR-104285 50-Mile Population Dose

<u>Calculate Peach Bottom Unit 2 NUREG/CR-4551 Study Population Dose.</u>

Given that the NUREG/CR-4551 does not document dose results as a function of collapsed APBs, calculations are performed to obtain these values. The calculation is as follows:

The Peach Bottom Unit 2 individual population dose (person-rem) for each collapsed APB is calculated by dividing the individual mean population dose (person-rem per reactor year) for each collapsed APB by the individual mean frequency contributions for each collapsed APB, or

$$PD_{CAPB} = MPD_{CAPB} \quad / \quad MF_{CAPB}$$

Where:
MPD_{CAPB} = individual mean population dose (person-rem per reactor year) for each collapsed APB
= $MFCR_{CAPB} * MPD_{50}$

Where:
$MFCR_{CAPB}$ = mean fractional contribution to risk for each collapsed APB [Section 2.3 Input#11]
MPD_{50} = mean population dose at 50 miles for Peach Bottom Unit 2 [Section 2.3 Input#8]

Therefore,

$$MPD_{CAPB1} = 0.0210 * 7.9 = 0.1659$$
$$MPD_{CAPB2} = 0.0066 * 7.9 = 0.05214$$
$$MPD_{CAPB3} = 0.5560 * 7.9 = 4.3924$$
$$MPD_{CAPB4} = 0.2260 * 7.9 = 1.7854$$
$$MPD_{CAPB5} = 0.0022 * 7.9 = 0.01738$$
$$MPD_{CAPB6} = 0.0590 * 7.9 = 0.4661$$
$$MPD_{CAPB7} = 0.1180 * 7.9 = 0.9322$$
$$MPD_{CAPB8} = 0.0005 * 7.9 = 0.00395$$
$$MPD_{CAPB9} = 0.0100 * 7.9 = 0.079$$
$$MPD_{CAPB10} = 0.000 * 7.9 = 0.0$$

Where:
MFC_{CAPB} = individual mean frequency contributions for each collapsed APB
= $IMFC_{CAPB} * CDF_{PBU2}$

Where:
$IMFC_{CAPB}$ = individual mean fractional contributions for each collapsed APB [Section 2.3 Input#12]

CDF_{PBU2} = Peach Bottom Unit 2 core damage frequency [Section 2.3 Input#9]

Therefore,

$$
\begin{aligned}
MFC_{CAPB1} &= 0.0220 * 4.34 \times 10^{-6} = 9.55 \times 10^{-8} \\
MFC_{CAPB2} &= 0.0110 * 4.34 \times 10^{-6} = 4.77 \times 10^{-8} \\
MFC_{CAPB3} &= 0.3410 * 4.34 \times 10^{-6} = 1.48 \times 10^{-6} \\
MFC_{CAPB4} &= 0.1830 * 4.34 \times 10^{-6} = 7.94 \times 10^{-7} \\
MFC_{CAPB5} &= 0.0030 * 4.34 \times 10^{-6} = 1.30 \times 10^{-8} \\
MFC_{CAPB6} &= 0.0470 * 4.34 \times 10^{-6} = 2.04 \times 10^{-7} \\
MFC_{CAPB7} &= 0.1100 * 4.34 \times 10^{-6} = 4.77 \times 10^{-7} \\
MFC_{CAPB8} &= 0.1840 * 4.34 \times 10^{-6} = 7.99 \times 10^{-7} \\
MFC_{CAPB9} &= 0.0890 * 4.34 \times 10^{-6} = 3.86 \times 10^{-7} \\
MFC_{CAPB10} &= 0.0100 * 4.34 \times 10^{-6} = 4.34 \times 10^{-8}
\end{aligned}
$$

Therefore,

P

$$
\begin{aligned}
D_{CAPB1} &= 0.1659 / 9.55 \times 10^{-8} = 1.74 \times 10^{6} \text{ (person-rem)} \\
PD_{CAPB2} &= 0.05214 / 4.77 \times 10^{-8} = 1.09 \times 10^{6} \text{ (person-rem)} \\
PD_{CAPB3} &= 4.3924 / 1.48 \times 10^{-6} = 2.97 \times 10^{6} \text{ (person-rem)} \\
PD_{CAPB4} &= 1.7854 / 7.94 \times 10^{-7} = 2.25 \times 10^{6} \text{ (person-rem)} \\
PD_{CAPB5} &= 0.01738 / 1.30 \times 10^{-8} = 1.34 \times 10^{6} \text{ (person-rem)} \\
PD_{CAPB6} &= 0.4661 / 2.04 \times 10^{-7} = 2.28 \times 10^{6} \text{ (person-rem)} \\
PD_{CAPB7} &= 0.9322 / 4.77 \times 10^{-7} = 1.95 \times 10^{6} \text{ (person-rem)} \\
PD_{CAPB8} &= 0.00395 / 7.99 \times 10^{-7} = 4.94 \times 10^{3} \text{ (person-rem)} \\
PD_{CAPB9} &= 0.079 / 3.86 \times 10^{-7} = 2.05 \times 10^{5} \text{ (person-rem)} \\
PD_{CAPB10} &= 0.0 / 4.34 \times 10^{-8} = 0.0 \text{ (person-rem)}
\end{aligned}
$$

Table 2-6 shows the 50-mile population dose (person-rem) for each APB considered in the NUREG/CR-4551 Peach Bottom Unit 2 study [12].

<u>Calculate FitzPatrick Person-Rem Per Collapsed Accident Progression Bin (APB).</u>

The Peach Bottom Unit [2] NUREG/CR-4551 consequences summarized in Table 2-6 (and calculated above) should be adjusted for use in this analysis to account for differences in the following parameters: reactor power level, technical specification allowed containment leakage rate, and population.

<u>Reactor Power Level Adjustment</u>

The adjustment factor for reactor power level is defined as the ratio of the power level at FitzPatrick to that at Peach Bottom Unit 2. This adjustment factor is calculated as follows:

$$AF_{power} = PLF \; / \; PLP$$

Where:

AF_{power} =th e adjustment factor for reactor power level
PLF = the power level at FitzPatrick = 2536 MWth [Section 2.3 Input #18]
PLP = the power level at Peach Bottom Unit 2 = 3293 MWth [Section 2.3 Input #15]

Therefore,

$$AF_{power} = 2536 / 3293 = 0.77$$

Technical Specification Allowed Containment Leakage Rate Adjustment

The adjustment factor for technical specification (TS) allowed containment leakage is defined as the ratio of the containment leakage at FitzPatrick to that at Peach Bottom Unit 2. This adjustment factor is calculated as follows:

$$AF_{Leakage} = LRF \;/\; LRP \qquad \text{\{equation 1\}}$$

Where:

$AF_{Leakage}$ = the adjustment factor for TS allowed containment leakage
LRF = the TS allowed containment leakage at FitzPatrick
LRP = the TS allowed containment leakage at Peach Bottom Unit 2

Because the leakage rates are in terms of the containment volume, the ratio of containment volumes is needed to relate the leakage rates. Therefore,

$$LRF = TS_{JAF} * VOL_{JAF} \qquad \text{\{equation 2\}}$$
$$LRP = TS_{PB} * VOL_{PB} \qquad \text{\{equation 3\}}$$

Where:

TS_{JAF}=	TS maximum allowed containment leakage is 1.5 volume%/day	[Section 2.3 Input #19]
VOL_{JAF} =	FitzPatrick containment free volume = 264, 000 ft^3	[Section 2,3 Input #20]
TS_{PB} =	TS maximum allowed containment leakage is 0.5 volume%/day	[Section 2.3 Input #16]
VOL_{PB} =	Peach Bottom Unit containment free volume = 307, 000 ft^3	[Section 2,3 Input #17]

Therefore, substituting equation 2 and 3 into 1 yields,

$$AF_{Leakage} = (1.5 * 264000) \;/\; (0.5 * 307000)$$
$$AF_{Leakage} = 2.58$$

Population Adjustment

The adjustment factor for population is defined as the ratio of the population within 50-mile radius of FitzPatrick to that of Peach Bottom Unit 2. This adjustment factor is calculated as follows:

$$AF_{Population} = POPF \;/\; POPP$$

Where:

$AF_{Population}$	=	the adjustment factor for population	
POPF	=	population within 50-mile radius of FitzPatrick = 8.98×10^5	[Section 2.3 Input #14]
POPP	=	population within 50-mile radius of Peach Bottom Unit 2	
	=	3.02×10^6	[Section 2.3 Input#13]

Therefore,

$$AF_{Population} = 8.98 \times 10^5 \;/\; 3.02 \times 10^6$$
$$AF_{Population} = 0.297$$

The above adjustment factors that are use in adjusting the population dose (person-rem) of the Peach Bottom Unit 2 for the FitzPatrick site and plant differences are as follows:

- Consequence categories dependent on the "INTACT" Tech Spec Leakage (collapsed accident progression bins 8 and 10)

$$AF_{8,10} = AF_{power} * AF_{Leakage} * AF_{Population}$$

Where:
$AF_{8,10}$ = adjustment factor for collapsed accident progression bins 8 and 10
AF_{power} = the adjustment factor for reactor power level = 0.77 [from above]
$AF_{Leakage}$ = the adjustment factor for TS allowed containment leakage = 2.58 [from above]
$AF_{Population}$ = the adjustment factor for population = 0.297 [from above]

Therefore,

$$AF_{8,10} = 0.77 * 2.58 * 0.297 = 0.59$$

- Consequence categories not dependent on the Technical Specification Leakage (collapsed accident progression bins 1, 2, 3, 4, 5, 6, 7 and 9)

$$AF_{1,2,3,4,5,6,7,9} = AF_{power} * AF_{Population}$$

Where:
$AF_{1,2,3,4,5,6,7,9}$ = adjustment factor for collapsed accident progression bins 1, 2, 3, 4, 5, 6, 7, and 9
AF_{power} = the adjustment factor for reactor power level = 0.77 [from above]
$AF_{Population}$ = the adjustment factor for population = 0.297 [from above]

Therefore,

$$AF_{1,2,3,4,5,6,7,9} = 0.77 * 0.297 = 0.22869$$

Based on the above adjustments factors for intact and non-intact accident progressions and the 50-mile population dose (person-rem) for each APB considered in the NUREG/CR-4551 Peach Bottom Unit 2 study, Table 2-6, the FitzPatrick doses (person-rem) are calculated as follows:

$$JAFMPD_{CAPB1} = AF_{1,2,3,4,5,6,7,9} * PD_{CAPB1}$$
$$JAFMPD_{CAPB2} = AF_{1,2,3,4,5,6,7,9} * PD_{CAPB2}$$
$$JAFMPD_{CAPB3} = AF_{1,2,3,4,5,6,7,9} * PD_{CAPB3}$$
$$JAFMPD_{CAPB4} = AF_{1,2,3,4,5,6,7,9} * PD_{CAPB4}$$
$$JAFMPD_{CAPB5} = AF_{1,2,3,4,5,6,7,9} * PD_{CAPB5}$$
$$JAFMPD_{CAPB6} = AF_{1,2,3,4,5,6,7,9} * PD_{CAPB6}$$
$$JAFMPD_{CAPB7} = AF_{1,2,3,4,5,6,7,9} * PD_{CAPB7}$$
$$JAFMPD_{CAPB8} = AF_{8,10} * PD_{CAPB8}$$
$$JAFMPD_{CAPB9} = AF_{1,2,3,4,5,6,7,9} * PD_{CAPB9}$$
$$JAFMPD_{CAPB10} = AF_{8,10} * PD_{CAPB10}$$

Therefore,

$$JAFMPD_{CAPB1} = 0.22869 * 1.74 \times 10^6 = 3.98 \times 10^5$$
$$JAFMPD_{CAPB2} = 0.22869 * 1.09 \times 10^6 = 2.49 \times 10^5$$
$$JAFMPD_{CAPB3} = 0.22869 * 2.97 \times 10^6 = 6.79 \times 10^5$$
$$JAFMPD_{CAPB4} = 0.22869 * 2.25 \times 10^6 = 5.15 \times 10^5$$
$$JAFMPD_{CAPB5} = 0.22869 * 1.34 \times 10^6 = 3.06 \times 10^5$$
$$JAFMPD_{CAPB6} = 0.22869 * 2.28 \times 10^6 = 5.21 \times 10^5$$
$$JAFMPD_{CAPB7} = 0.22869 * 1.95 \times 10^6 = 4.46 \times 10^5$$

$JAFMPD_{CAPB8}$ = 0.59 * 4.94×10^3 = 2.91×10^3
$JAFMPD_{CAPB9}$ = 0.22869 * 2.05×10^5 = 4.69×10^4
$JAFMPD_{CAPB10}$ = 0.59 * 0.0 = 0.0

Table 2-7 summarizes the Peach Bottom Unit 2 NUREG/CR-4551 [12] doses after adjustment for changes in population, reactor power level, and containment leakage rate for application to FitzPatrick.

Calculate FitzPatrick Accident Progression Bin (APB) Frequency

The FitzPatrick person-rem frequency is calculated in terms of collapsed accident progression bins (Table 2-4). The calculation is performed by running the FitzPatrick Level 2 containment event tree model to match the criteria for the Peach Bottom Unit 2 NUREG/CR-4551 study [12]. The results of this sort is presented in Table 2-8 and detailed in Attachment A.

Calculate EPRI 's TR-104285 50-Mile Population Dose

The FitzPatrick's person-rem results (Table 2-7) are converted to match to the EPRI TR-104285 release classes. The calculation assigns each of the FitzPatrick Level 2 source term category endstates (Tables 2-2 and 2-3) to the equivalent NUREG/CR-4551 Peach Bottom Unit 2 [12] collapsed accident progression bin category (Table 2-4).

This is required because the FitzPatrick IPE Level 2 results are not defined in the same terms as reported in NUREG/CR-4551. Therefore, in order to use the Level 3 results presented in NUREG/CR-4551 for Peach Bottom Unit 2 [12], the FitzPatrick IPE Level 2 results needs to be converted into a format that allows the use of the Peach Bottom Unit Level 3 results.

The FitzPatrick IPE provides a grouping of containment failure modes and subsequent release categories resulting from severe accident challenges. Tables 2-2 and 2-3 provide this breakdown for the FitzPatrick IPE. The FitzPatrick release endstates of Tables 2-2 and 2-3 were reviewed and assigned into one of the collapsed accident progression bins from NUREG/CR-4551 [12]. The result of this review is presented in Table 2-9.

The following discussion provides the basis for the assignment of population dose for each EPRI accident class.

The 50-miles population dose for the EPRI accident class "no containment failure" is based on the FitzPatrick's collapsed accident progression bins 8 and 10 (Table 2-7) as the ones closest to the definition of an intact containment. The population dose is calculated as the weighted mean 50-mile population dose for each collapsed accident progression bin.

$$CLASS\_1\_DOSE = JAFWPD_{CAPB8} + JAFWPD_{CAPB10}$$

Where:
CLASS_1_DOSE = 50-miles population dose for the EPRI accident class "no containment failure"
$JAFWPD_{CAPB8}$ = FitzPatrick 50-miles collapsed APB 8 weighted mean population dose (person-rem)
$JAFWPD_{CAPB10}$ = FitzPatrick 50-miles collapsed APB 10 weighted mean population dose (person-rem)

$$JAFWPD_{CAPB8} = \left[\frac{JAF\text{-}F_{CAPB8}}{JAF\text{-}F_{CAPB8} + JAF\text{-}F_{CAPB10}}\right] * JAFMPD_{CAPB8}$$

$$JAFWPD_{CAPB10} = \left[\frac{JAF\text{-}F_{CAPB10}}{JAF\text{-}F_{CAPB8} + JAF\text{-}F_{CAPB10}}\right] * JAFMPD_{CAPB10}$$

Where:
$JAF\text{-}F_{CAPB8}$ = FitzPatrick collapsed APB 8 mean frequency [Table 2-8]
$JAF\text{-}F_{CAPB10}$ = FitzPatrick collapsed APB 10 mean frequency [Table 2-8]
$JAFMPD_{CAPB8}$ = FitzPatrick 50-miles collapsed APB 8 mean population dose (person-rem) [Table 2-7]
$JAFMPD_{CAPB10}$ = FitzPatrick 50-miles collapsed APB 10 mean population dose (person-rem) [Table 2-7]

Therefore, CLASS_1_DOSE = 2.91×10^3 person-rem

The 50-miles population dose for the EPRI accident Class 2 (Large Containment Isolation Failures, failure-to-close) is based on the FitzPatrick's collapsed accident progression bin 3 (Table 2-7) 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 3 results in the highest dose of all of the FitzPatrick "containment failure" collapsed accident progression bins (which is indicative of a containment failure with torus pool and drywell bypass).

Therefore, CLASS_2_DOSE = 6.79×10^5 person-rem

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 [3], are taken as factors of 10La and 35La [4, 8], respectively, times the population dose of EPRI accident Class 1.

Therefore,

CLASS_3a_DOSE = 10 * CLASS_1_DOSE
CLASS_3b_DOSE = 35 * CLASS_1_DOSE

CLASS_3a_DOSE = $10 * 2.91 \times 10^3$
CLASS_3b_DOSE = $35 * 2.91 \times 10^3$

CLASS_3a_DOSE = 2.91×10^4 person-rem
CLASS_3b_DOSE = 1.02×10^5 person-rem

Per the NEI Interim Guidance [3], 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.

The 50-miles population dose for the EPRI accident Class 7a (Severe Accident Phenomena Induced Early Drywell Failures) is based on the FitzPatrick's collapsed accident progression bins 3 and 4 (Table 2-7) as the ones closest to the definition of early drywell failures. The population dose is calculated as the weighted mean 50-mile population dose for each collapsed accident progression bin.

CLASS_7a_DOSE = $JAFWPD_{CAPB3} + JAFWPD_{CAPB4}$

Where:

CLASS_7a_DOSE = 50-miles population dose for the EPRI accident class 7a
$JAFWPD_{CAPB3}$ = FitzPatrick 50-miles collapsed APB 3 weighted mean population dose (person-rem)
$JAFWPD_{CAPB4}$ = FitzPatrick 50-miles collapsed APB 4 weighted mean population dose (person-rem)

$$JAFWPD_{CAPB3} = \left[\frac{JAF\text{-}F_{CAPB3}}{JAF\text{-}F_{CAPB3} + JAF\text{-}F_{CAPB4}}\right] * JAFMPD_{CAPB3}$$

$$JAFWPD_{CAPB4} = \left[\frac{JAF\text{-}F_{CAPB4}}{JAF\text{-}F_{CAPB3} + JAF\text{-}F_{CAPB4}}\right] * JAFMPD_{CAPB4}$$

Where:
$JAF\text{-}F_{CAPB3}$ = FitzPatrick collapsed APB 3 mean frequency [Table 2-8]
$JAF\text{-}F_{CAPB4}$ = FitzPatrick collapsed APB 4 mean frequency [Table 2-8]
$JAFMPD_{CAPB3}$ = FitzPatrick 50-miles collapsed APB 3 mean population dose (person-rem) [Table 2-7]
$JAFMPD_{CAPB4}$ = FitzPatrick 50-miles collapsed APB 4 mean population dose (person-rem) [Table 2-7]

Therefore, CLASS_7a_DOSE = 6.11×10^5 person-rem

The 50-miles population dose for the EPRI accident Class 7b (Severe Accident Phenomena Induced Early Torus Failures) is based on the FitzPatrick's collapsed accident progression bins 1 and 2 (Table 7) as the ones closest to the definition of early torus failures. The population dose is calculated as the weighted mean 50-mile population dose for each collapsed accident progression bin.

$$CLASS\_7b\_DOSE = JAFWPD_{CAPB1} + JAFWPD_{CAPB2}$$

Where:
CLASS_7b_DOSE = 50-miles population dose for the EPRI accident class 7b
$JAFWPD_{CAPB1}$ = FitzPatrick 50-miles collapsed APB 1 weighted mean population dose (person-rem)
$JAFWPD_{CAPB2}$ = FitzPatrick 50-miles collapsed APB 2 weighted mean population dose (person-rem)

$$JAFWPD_{CAPB1} = \left[\frac{JAF\text{-}F_{CAPB1}}{JAF\text{-}F_{CAPB1} + JAF\text{-}F_{CAPB2}}\right] * JAFMPD_{CAPB1}$$

$$JAFWPD_{CAPB2} = \left[\frac{JAF\text{-}F_{CAPB2}}{JAF\text{-}F_{CAPB1} + JAF\text{-}F_{CAPB2}}\right] * JAFMPD_{CAPB2}$$

Where:
$JAF\text{-}F_{CAPB1}$ = FitzPatrick collapsed APB 1 mean frequency [Table 2-8]
$JAF\text{-}F_{CAPB2}$ = FitzPatrick collapsed APB 2 mean frequency [Table 2-8]
$JAFMPD_{CAPB1}$ = FitzPatrick 50-miles collapsed APB 1 mean population dose (person-rem) [Table 2-7]
$JAFMPD_{CAPB2}$ = FitzPatrick 50-miles collapsed APB 2 mean population dose (person-rem) [Table 2-7]

Therefore, CLASS_7b_DOSE = 3.96×10^5 person-rem

The 50-miles population dose for the EPRI accident Class 7c (Severe Accident Phenomena Induced Late Drywell Failures) is based on the FitzPatrick's collapsed accident progression bin 6 (Table 2-7) as the one closest to the definition of late drywell failures.

Therefore, CLASS_7c_DOSE = 5.21×10^5 person-rem

The 50-miles population dose for the EPRI accident Class 7d (Severe Accident Phenomena Induced Late Torus Failures) is based on the FitzPatrick's collapsed accident progression bin 5 (Table 2-7) as the one closest to the definition of late torus failures.

Therefore, CLASS_7d_DOSE = 3.06×10^5 person-rem

The 50-miles population dose for the EPRI accident Class 8 (Bypass) is based on the FitzPatrick's collapsed accident progression bin 7 (Table 2-7) as the one closest to the definition of bypass failure. This selection is based the dominance of ATWS induced bypass failure of the torus for this accident class. Collapsed accident progression bin 7 represents the release due to torus venting, therefore, it 's indicative of containment bypass scenarios.

Therefore, CLASS_8_DOSE = 4.46×10^5 person-rem

Using the preceding information, the population dose for the 50-mile radius surrounding Fitzpatrick is summarized in Table 2-10. (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 [3]).

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 2-10) 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

Therefore,

$CLASS\_1\_DOSE_{RATE} = 2.91 \times 10^3 * 6.19 \times 10^{-7} = 1.80 \times 10^{-3}$ (person-rem/ry)
$CLASS\_2\_DOSE_{RATE} = 6.79 \times 10^5 * 2.44 \times 10^{-9} = 1.66 \times 10^{-3}$ (person-rem/ry)
$CLASS\_3a\_DOSE_{RATE} = 2.91 \times 10^4 * 6.59 \times 10^{-8} = 1.92 \times 10^{-3}$ (person-rem/ry)
$CLASS\_3b\_DOSE_{RATE} = 1.02 \times 10^5 * 6.59 \times 10^{-9} = 6.72 \times 10^{-4}$ (person-rem/ry)
$CLASS\_7a\_DOSE_{RATE} = 6.11 \times 10^5 * 1.74 \times 10^{-7} = 1.06 \times 10^{-1}$ (person-rem/ry)

$CLASS\_7b\_DOSE_{RATE}$	=	3.96×10^{5}	*	7.37×10^{-7}	=	2.92×10^{-1} (person-rem/ry)
$CLASS\_7c\_DOSE_{RATE}$	=	5.21×10^{5}	*	5.82×10^{-8}	=	3.03×10^{-2} (person-rem/ry)
$CLASS\_7e\_DOSE_{RATE}$	=	3.06×10^{5}	*	4.83×10^{-7}	=	1.48×10^{-1} (person-rem/ry)
$CLASS\_8\_DOSE_{RATE}$	=	4.46×10^{5}	*	2.97×10^{-7}	=	1.32×10^{-1} (person-rem/ry)

Table 2-11 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 [7] and per the NEI Interim Guidance [3], 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[8], 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 "Data 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 FitzPatrick 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. [17]. 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 FitzPatrick ILRT testing frequency. This is performed as follows:

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

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

Where:

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

$PROB_{class\_3b\_10}$ = probability of large 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

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

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]

$SURTEST_{10}$ = surveillance interval of interest, months/2 = 10 years*12 months/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 FitzPatrick 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.

$PROB_{class\_3b\_15}$ = probability of large pre-existing containment liner leakage given a 1-in-15 years ILRT frequency.

$SURTEST_{15}$ = surveillance interval of interest, months/2 = 15 years*12 months/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.

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

$$CLASS\_3a\_FREQUENCY_{10} = PROB_{class\_3a\_10} * CDF$$

$$CLASS\_3a\_FREQUENCY_{15} = PROB_{class\_3a\_15} * CDF$$

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 = FitzPatrick IPE core damage frequency = 2.44×10^{-6}/ry [Section 2.3, input#2]

Therefore,

$$CLASS\_3a\_FREQUENCY_{10} = 0.090 * 2.44 \times 10^{-6} = 2.20 \times 10^{-7}/ry$$
$$CLASS\_3a\_FREQUENCY_{15} = 0.135 * 2.44 \times 10^{-6} = 3.29 \times 10^{-7}/ry$$

<u>Frequency of EPRI Class 3b Sequences.</u> Consistent with NEI Interim Guidance [3], 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

Therefore,

$$CLASS\_3b\_FREQUENCY_{10} = 0.0090 * 2.44 \times 10^{-6} = 2.20 \times 10^{-8}/ry$$
$$CLASS\_3b\_FREQUENCY_{15} = 0.0135 * 2.44 \times 10^{-6} = 3.29 \times 10^{-8}/ry$$

<u>Frequency of EPRI Class 1 Sequences.</u> Consistent with NEI Interim Guidance [3], 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:

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

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

Where:

NCF = frequency in which containment leakage is at or below maximum allowable Technical Specification Leakage = 6.91×10^{-6} /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

Therefore:

$$CLASS\_1\_FREQUENCY_{10} = 6.91 \times 10^{-7} - 2.20 \times 10^{-7} - 2.20 \times 10^{-8} = 4.49 \times 10^{-7}/ry$$

$$CLASS\_1\_FREQUENCY_{15} = 6.91 \times 10^{-7} - 3.29 \times 10^{-7} - 3.29 \times 10^{-8} = 3.29 \times 10^{-7}/ry$$

The impacted frequencies of the EPRI accident classes are summarized in Table 2-12.

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

This step, per the NEI Interim Guidance [3], calculates the population dose rate for the new surveillance intervals of interest by multiplying the population dose (Table 2-10) by the frequency for each of the eight EPRI's accident classes (Tables 2-5 and 2-12). In addition, sum the accident class dose rates to obtain the total dose rate.

Per the NEI Interim Guidance [3], 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:

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

Where:

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

$CLASS\_7d\_DOSE_{RATE-10}$	=	EPRI accident Class 7d dose rate given a 1-in-10 years ILRT interval
$CLASS\_8\_DOSE_{RATE-10}$	=	EPRI accident Class 8 dose rate given a 1-in-10 years ILRT interval

Therefore,

$CLASS\_1\_DOSE_{RATE-10}$	=	2.91×10^{3}	*	4.49×10^{-7}	=	1.31×10^{-3} (person-rem/ry)
$CLASS\_2\_DOSE_{RATE-10}$	=	6.79×10^{5}	*	2.44×10^{-9}	=	1.66×10^{-3} (person-rem/ry)
$CLASS\_3a\_DOSE_{RATE-10}$	=	2.91×10^{4}	*	2.20×10^{-7}	=	6.40×10^{-3} (person-rem/ry)
$CLASS\_3b\_DOSE_{RATE-10}$	=	1.02×10^{5}	*	2.20×10^{-8}	=	2.24×10^{-3} (person-rem/ry)
$CLASS\_7a\_DOSE_{RATE-10}$	=	6.11×10^{5}	*	1.74×10^{-7}	=	1.06×10^{-1} (person-rem/ry)
$CLASS\_7b\_DOSE_{RATE-10}$	=	3.96×10^{5}	*	7.37×10^{-7}	=	2.92×10^{-1} (person-rem/ry)
$CLASS\_7c\_DOSE_{RATE-10}$	=	5.21×10^{5}	*	5.82×10^{-8}	=	3.03×10^{-2} (person-rem/ry)
$CLASS\_7d\_DOSE_{RATE-10}$	=	3.06×10^{5}	*	4.83×10^{-7}	=	1.48×10^{-1} (person-rem/ry)
$CLASS\_8\_DOSE_{RATE-10}$	=	4.46×10^{5}	*	2.97×10^{-7}	=	1.32×10^{-1} (person-rem/ry)

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

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

Where:

$CLASS\_1\_DOSE_{RATE-15}$	=	EPRI accident Class 1 dose rate given a 1-in-15 years ILRT interval
$CLASS\_2\_DOSE_{RATE-15}$	=	EPRI accident Class 2 dose rate given a 1-in-15 years ILRT interval
$CLASS\_3a\_DOSE_{RATE-15}$	=	EPRI accident Class 3a dose rate given a 1-in-15 years ILRT interval
$CLASS\_3b\_DOSE_{RATE-15}$	=	EPRI accident Class 3b dose rate given a 1-in-15 years ILRT interval
$CLASS\_7a\_DOSE_{RATE-15}$	=	EPRI accident Class 7a dose rate given a 1-in-15 years ILRT interval
$CLASS\_7b\_DOSE_{RATE-15}$	=	EPRI accident Class 7b dose rate given a 1-in-15 years ILRT interval
$CLASS\_7c\_DOSE_{RATE-15}$	=	EPRI accident Class 7c dose rate given a 1-in-15 years ILRT interval
$CLASS\_7d\_DOSE_{RATE-15}$	=	EPRI accident Class 7d dose rate given a 1-in-15 years ILRT interval
$CLASS\_8\_DOSE_{RATE-15}$	=	EPRI accident Class 8 dose rate given a 1-in-15 years ILRT interval

Therefore,

$CLASS\_1\_DOSE_{RATE-15}$	=	2.91×10^{3}	*	3.29×10^{-7}	=	9.58×10^{-4} (person-rem/ry)
$CLASS\_2\_DOSE_{RATE-15}$	=	6.79×10^{5}	*	2.44×10^{-9}	=	1.66×10^{-3} (person-rem/ry)
$CLASS\_3a\_DOSE_{RATE-15}$	=	2.91×10^{4}	*	3.29×10^{-7}	=	9.60×10^{-3} (person-rem/ry)
$CLASS\_3b\_DOSE_{RATE-15}$	=	1.02×10^{5}	*	3.29×10^{-8}	=	3.36×10^{-3} (person-rem/ry)
$CLASS\_7a\_DOSE_{RATE-15}$	=	6.11×10^{5}	*	1.74×10^{-7}	=	1.06×10^{-1} (person-rem/ry)
$CLASS\_7b\_DOSE_{RATE-15}$	=	3.96×10^{5}	*	7.37×10^{-7}	=	2.92×10^{-1} (person-rem/ry)
$CLASS\_7c\_DOSE_{RATE-15}$	=	5.21×10^{5}	*	5.82×10^{-8}	=	3.03×10^{-2} (person-rem/ry)
$CLASS\_7d\_DOSE_{RATE-15}$	=	3.06×10^{5}	*	4.83×10^{-7}	=	1.48×10^{-1} (person-rem/ry)
$CLASS\_8\_DOSE_{RATE-15}$	=	4.46×10^{5}	*	2.97×10^{-7}	=	1.32×10^{-1} (person-rem/ry)

The dose rates per EPRI accident class as a function of ILRT interval are summarized in Table 2-13.

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

This step, per the NEI Interim Guidance [3] 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 2-13, for the current FitzPatrick 1-in10 years ILRT interval, the percentage contribution to total dose rate from EPRI's accident Classes 3a and 3b is calculated as follows:

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

Where:

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

$TOT\text{-}DOSE_{RATE\text{-}10}$ = Total dose rate for all EPRI's Classes given a 1-in-10 years ILRT interval
= 0.720 [Table 2-13]

Therefore,

$$PER\_CHG_{10} = \left[\frac{6.40 \times 10^{-3} + 2.24 \times 10^{-3}}{0.720}\right] * 100$$

PER_CHG_{10} = 1.2%

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:

PER_CHG_{15} = percentage contribution to total dose rate from EPRI's accident Classes 3a and 3b given a 1-in-15 years ILRT interval

$TOT\text{-}DOSE_{RATE\text{-}15}$ = Total dose rate for all EPRI's Classes given a 1-in-15 years ILRT interval
= 0.724 (person-rem/ry) [Table 2-13]

Therefore,

$$PER\_CHG_{15} = \left[\frac{9.60 \times 10^{-3} + 3.36 \times 10^{-3}}{0.724}\right] * 100$$

PER_CHG_{15} = 1.8%

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

Therefore,

$$INCREASE_{10\text{-}15} = \left[\frac{0.724 - 0.720}{0.720}\right] * 100 = 0.56\%$$

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 0.56%. 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 [3] 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 FitzPatrick IPE [6], 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 2-12) is used as the LERF for FitzPatrick.

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

$$\Delta LERF_{10\text{-}15} = CLASS\_3b\_FREQUENCY_{15} - CLASS\_3b\_FREQUENCY_{10}$$

Where:

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

$CLASS\_3b\_FREQUENCY_{15}$ = frequency of EPRI accident Class 3b given a 1-in-15 years ILRT Interval = 3.29×10^{-8} /ry [Table 2-12]

CLASS_3b_FREQUENC Y_{10} = frequency of EPRI accident Class 3b given a 1-in-10 years ILRT Interval = 2.20×10^{-8}/ry [Table 2-12]

Therefore,

$$\Delta LERF_{10\text{-}15} = 3.29 \times 10^{-8} - 2.20 \times 10^{-8}$$

$$\Delta LERF_{10\text{-}15} = 1.09 \times 10^{-8}/ry$$

Regulatory Guide 1.174 [5] 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 ΔLERF of 1.09×10^{-8}/ry falls into Region III, Very Small Change in Risk (Figure 2-1), of the acceptance guidelines in NRC Regulatory Guide 1.174 [5]. Therefore, because Regulatory Guide 1.174 [5] defines very small changes in LERF as below 10^{-7}/yr, increasing the ILRT interval at FitzPatrick 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:

$$\Delta LERF_{3\text{-}15} = \text{CLASS\_3b\_FREQUENC } Y_{15} - \text{CLASS\_3b\_FREQUENC } Y_{3}$$

Where:

$\Delta LERF_{3\text{-}15}$ = 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 = 3.29×10^{-8}/ry [Table 2-12]

CLASS_3b_FREQUENC Y_{3} = frequency of EPRI accident Class 3b given a 1-in-10 years ILRT Interval = 6.59×10^{-9}/ry [Table 2-12]

Therefore,

$$\Delta LERF_{3\text{-}15} = 3.29 \times 10^{-8} - 6.59 \times 10^{-9}$$

$$\Delta LERF_{3\text{-}15} = 2.63 \times 10^{-8}/ry$$

Similar to the $\Delta LERF_{10\text{-}15}$ result, the $\Delta LERF_{3\text{-}15}$ 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 [3] 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 expose in Regulatory Guide 1.174 [5][9].

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

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 [3], 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 = FitzPatrick IPE core damage frequency = 2.44×10^{-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 = 4.49×10^{-7}/ry [Table 2-12]

$CLASS\_3a\_FREQUENC\,Y_{10}$ = frequency of EPRI accident Class 3a given a 1-in-10 years ILRT Interval = 2.20×10^{-7}/ry [Table 2-12]

Therefore,

$$CCFP_{10} = \left\{1 - \left(\left[\frac{4.49 \times 10^{-7} + 2.20 \times 10^{-7}}{2.44 \times 10^{-6}}\right]\right)\right\} * 100\%$$

$CCFP_{10}$ = 72.6%

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 = FitzPatrick IPE core damage frequency = 2.44×10^{-6}/ry [Section 2.3, input#2]

CLASS_1_FREQUENC Y_{15} = frequency of EPRI accident Class 1 given a 1-in-15 years ILRT Interval = 3.29×10^{-7}/ry [Table 12]

CLASS_3a_FREQUENC Y_{15} = frequency of EPRI accident Class 3a given a 1-in-15 years ILRT Interval = 3.29×10^{-7}/ry [Table 12]

Therefore,

$$CCFP_{15} = \left\{1 - \left(\left[\frac{3.29 \times 10^{-7} + 3.29 \times 10^{-7}}{2.44 \times 10^{-6}}\right]\right)\right\} * 100\%$$

$CCFP_{15}$ = 73.03%

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

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

$$\Delta CCFP_{10\text{-}15} = 73.03\% - 72.6\%$$

$$\Delta CCFP_{10\text{-}15} = 0.43\%$$

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 [19], FitzPatrick submitted an Individual Plant Examination of External Events (IPEEE) in June 1996 [20]. 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 FitzPatrick 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 FitzPatrick IPEEE submittal [20] examined a spectrum of external events hazards based on acceptable screening methods (NRC seismic margin [21, 22], EPRI Fire PRA methodology [23], 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 FitzPatrick 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 1.03 x 10^{-7}/ry. This delta LERF is just slightly above the Region III boundary for LERF (Figure 2-1) and falls within NRC Regulatory Guide 1.174 [5] Region II ("Small Changes" in risk). As stated above, this can be attributed to the conservative screening nature of the external event methods available for their quantitative assessment at FitzPatrick.

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 0.16% or 0.038 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 0.20%. 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 FitzPatrick ILRT interval from 10 to 15 years is an acceptable plant change from a risk perspective). This results is 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.

<u>**Request for Additional Information:**</u>

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 [24] 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 FitzPatrick 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 [24], 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.

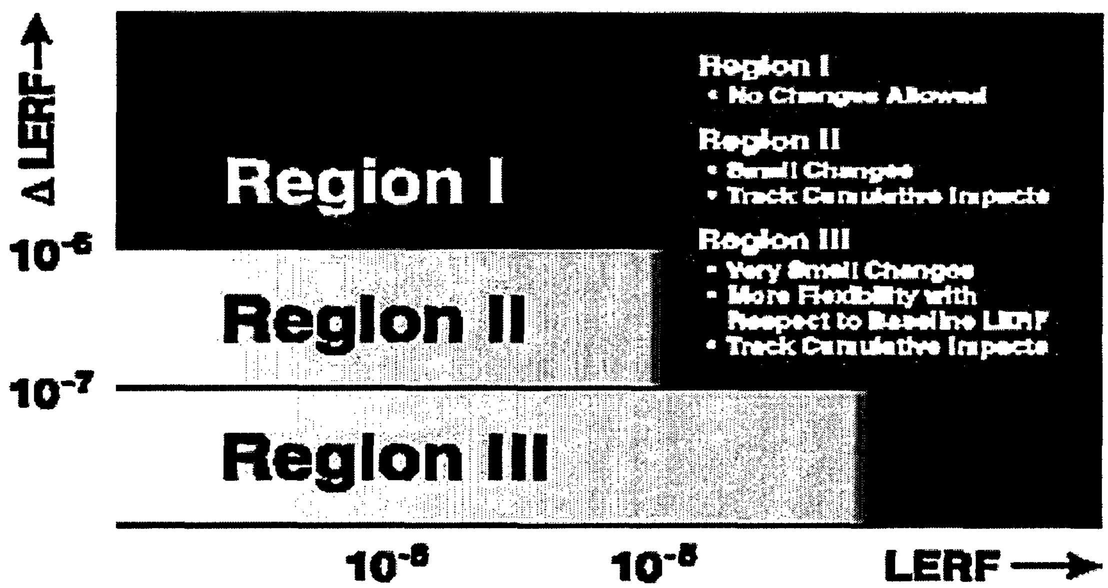
9) Evaluate the change in conditional containment failure probability.

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 1.22×10^{-8}/ry. This value is below the NRC Regulatory Guide 1.174 [5] 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 FitzPatrick 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 4.1×10^{-3} person-rem/ry or 0.57%, and the conditional containment failure probability increase is estimated to be 0.6%. 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[10] for Large Early Release Frequency [5]



[10] 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

Internal Core Damage Frequency Contributions by Plant Damage States

PDS	Simplified Description	Point Estimate PDS Freq (/ry)	% Of Total CDF
1	Short-term large LOCA with loss of core cooling. Core damage results at low reactor pressure. Late injection and containment heat removal is available.	1.44×10^{-8}	0.59
2	Short-term medium size LOCA with loss of core cooling. Core damage results at reactor pressure. Late injection, and containment heat removal are available.	1.90×10^{-7}	7.78
3	Short-term transient with loss of core cooling. Core damage results at high reactor pressure. Late injection, and containment heat removal are available.	1.25×10^{-6}	51.10
4	Long-term SBO with battery depletion. Core damage results at high reactor pressure. All accident-mitigating functions are recoverable when ac power is restored.	2.13×10^{-7}	8.75
5	Short-term SBO with no dc power. Core damage results at high reactor pressure. All accident-mitigating functions are recoverable when ac power is restored.	4.58×10^{-8}	1.88
6	Long-term SBO involving a loss of high-pressure injection because of one stuck-open safety relief valve. Core damage results at low reactor pressure. All accident-mitigating functions are recoverable when ac power is restored.	8.66×10^{-8}	3.55
7	Short-term SBO involving a loss of high-pressure injection because of two stuck-open safety relief valves. Core damage results at low reactor pressure. All accident mitigating functions are recoverable when ac power is restored	6.38×10^{-9}	0.26
8	Short-term transient with loss of core cooling. Core damage results at low reactor pressure. Late injection, and containment heat removal are available.	4.00×10^{-8}	1.64
9	Short-term ATWS with one stuck-open safety relief valve that leads to early core damage at low reactor pressure following loss of reactivity control. Late injection and containment heat removal is available.	1.62×10^{-8}	0.66
10	Short-term ATWS that leads to early core damage at high reactor pressure following loss of reactivity control. Late injection and containment heat removal is available.	2.81×10^{-7}	11.51
11	Transient with a loss of long-term decay heat removal. Core damage results at high reactor pressure. Late in-vessel and ex-vessel injection is available.	3.00×10^{-7}	12.29
Total		2.44×10^{-6}	100

Table 2-2

Summary of FitzPatrick IPE LEVEL 2 Release Categories [6]

Release Category	Description	Frequency (/ry)	% Of Total CDF
Early Low	NA	0.00	0.00
Early Medium Low	Small early torus failures [11]	5.04×10^{-7}	20.66
Early Medium High	Large early torus failures or early drywell failures with no core-concrete interactions (drywell flooded)	5.29×10^{-7}	21.70
Early High	Early drywell failure with no drywell sprays	1.74×10^{-7}	7.14
Late Low	Vessel breach, no containment failure and no core-concrete interactions	9.76×10^{-9}	0.004
Late Medium Low	Small late torus failures	4.73×10^{-7}	19.40
Late Medium High	Large late torus failures	2.71×10^{-11}	0.00001
Late High	Late drywell failure with no drywell sprays	5.82×10^{-8}	2.39
No Containment Failure	Containment Intact (no release)	6.91×10^{-7}	28.34
Total		2.44×10^{-6}	100

Table 2-3

Summary of FitzPatrick IPE LEVEL 2 Containment Failures [6]

Containment End State	Release Category	Frequency (/ry)	% Of Total CDF
No Containment Failure	No Containment Failure	6.91×10^{-7}	28.34
Early Drywell Failure	Early High	1.74×10^{-7}	7.14
Early Wetwell Failure	Non-ATWS Early Medium Low, Non-ATWS Early Medium High	7.37×10^{-7}	30.19
Late Drywell Failure	Late High	5.82×10^{-8}	2.39
Late Wetwell Failure	Late Low, Late Medium Low, Late Medium High	4.83×10^{-7}	19.80
Bypass [12]	ATWS Early Medium Low, ATWS Early Medium High	2.97×10^{-7}	12.17
Total		2.44×10^{-6}	100

[11] Includes torus venting.
[12] Only ATWS bypass sequences considered, ISLOCA sequences are negligible.

Table 2-4

Collapsed Accident Progression Bins (APB) Descriptions

Collapsed APB Number	Description
CAPB-1	[CD, VB, Early CF, WW, RPV pressure >200 psig at VB] 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).
CAPB-2	[CD, VB, Early CF, WW, RPV pressure <200 psig at VB] 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.
CAPB-3	[CD, VB, Early CF, DW, RPV pressure >200 psig at VB] 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).
CAPB-4	[CD, VB, Early CF, DW, RPV pressure <200 psig at VB] 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).
CAPB-5	[CD, VB, Late CF, WW] 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.

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

Table 2-4

Collapsed Accident Progression Bins (APB) Descriptions (continued)

Collapsed APB Number	Description
CAPB-6	[CD, VB, Late CF, DW] 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 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.
CAPB-7	[CD, VB, Vent] Core damage (CD) occurs followed by vessel breach (VB). The containment fails because of venting during any of the time periods in the accident.
CAPB-8	[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.
CAPB-9	[CD, No VB] Core damage occurs (CD), but the recovery of RPV injection in time prevents vessel beach (No VB). However, the containment can fail even if vessel breach is averted. Therefore, the potential exists for some of the in-vessel releases to be released to the environment.
CAPB-10	[No CD] Core damage does not occur (No CD). Therefore, containment integrity is not challenged (No CF).

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 FitzPatrick 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 accident classes' 3a and 3b.	6.19×10^{-7}
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.	2.44×10^{-9}
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.	6.59×10^{-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.	6.59×10^{-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.	1.74×10^{-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.	7.37×10^{-7}
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.	5.82×10^{-8}
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.	4.83×10^{-7}
8	Containment Bypassed (ATWS)	Accident sequences in which the containment is bypassed (i.e., ATWS with high power oscillations or Interfacing Systems LOCA, ISLOCA).	2.97×10^{-7}
CDF	All Level 2 CET Endstates		2.44×10^{-6}

Table 2-6
Peach Bottom Unit 2 NUREG/CR-4551 Collapsed Accident Progression Bin 50-Mile Population Dose

APB #	Fractional APB Contributions[1]	PBAPS Population Dose Risk (Person-rem/ry)[2]	PBAPS APB Frequencies (1/ry)[3]	PBAPS Population Dose (Person-rem)[4]
1	0.021	0.1659	9.55×10^{-8}	1.74×10^{6}
2	0.0066	0.05214	4.77×10^{-8}	1.09×10^{6}
3	0.556	4.3924	1.48×10^{-6}	2.97×10^{6}
4	0.226	1.7854	7.94×10^{-7}	2.25×10^{6}
5	0.0022	0.01738	1.30×10^{-8}	1.34×10^{6}
6	0.059	0.4661	2.04×10^{-7}	2.28×10^{6}
7	0.118	0.9322	4.77×10^{-7}	1.95×10^{6}
8	0.0005	0.00395	7.99×10^{-7}	4.94×10^{3}
9	0.01	0.079	3.86×10^{-7}	2.05×10^{5}
10	0	0	4.34×10^{-7}	0.0
Totals	1	7.9	4.34×10^{-6}	1.38×10^{7}

1. Obtained from table 5.2-3 of NUREG/CR-4551 [12]
2. Derived from the fractional APB contributions times total population dose risk, 0.021 x 7.9 = 0.1659
3. Derived from the conditional probabilities of the APBs and the total internal CDF given in NUREG/CR-4551 [12] (See Step 2.3.3)
4. $0.1659 / 9.55 \times 10^{-8} = 1.74 \times 10^{6}$

Table 2-7

FitzPatrick Population Doses for the Each APB and Associated Adjustment Factors

PBAPS APB #	PBAPS Population Dose (Person-rem)	Adjustment Factors			JAF Population Dose (Person-rem)
		Population	Reactor Power Level	TS Allowed Containment Leakage Rate	
1	1.74×10^{6}	0.297	0.770	NA	3.98×10^{5}
2	1.09×10^{6}	0.297	0.770	NA	2.49×10^{5}
3	2.97×10^{6}	0.297	0.770	NA	6.79×10^{5}
4	2.25×10^{6}	0.297	0.770	NA	5.15×10^{5}
5	1.34×10^{6}	0.297	0.770	NA	3.06×10^{5}
6	2.28×10^{6}	0.297	0.770	NA	5.21×10^{5}
7	1.95×10^{6}	0.297	0.770	NA	4.46×10^{5}
8	4.94×10^{3}	0.297	0.770	2.58	2.91×10^{3}
9	2.05×10^{5}	0.297	0.770	NA	4.69×10^{4}
10	0.0	0.297	0.770	2.58	0.0

Table 2-8

FitzPatrick Collapsed Accident Progression Bins Frequencies

Collapsed APB	Description	Frequency (Ry)
CAPB-1	[CD, VB, Early CF, WW, RPV pressure >200 psig at VB] 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).	1.43×10^{-7}
CAPB-2	[CD, VB, Early CF, WW, RPV pressure <200 psig at VB] 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 due to the present of an overlying pool of water.	2.27×10^{-9}
CAPB-3	[CD, VB, Early CF, DW, RPV pressure >200 psig at VB] 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).	5.41×10^{-7}
CAPB-4	[CD, VB, Early CF, DW, RPV pressure <200 psig at VB] 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).	3.86×10^{-7}
CAPB-5	[CD, VB, Late CF, WW] 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.	3.09×10^{-10}

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

Table 2-8

FitzPatrick Collapsed Accident Progression Bins Frequencies (continued)

Collapsed APB	Description	Frequency (Ry)
CAPB-6	[CD, VB, Late CF, DW] 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 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.	1.31×10^{-7}
CAPB-7	[CD, VB, Vent] Core damage (CD) occurs followed by vessel breach (VB). The containment fails because of venting during any of the time periods in the accident.	5.45×10^{-7}
CAPB-8	[CD, VB, No CF] 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.	2.92×10^{-7}
CAPB-9	[CD, No VB] 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.	4.00×10^{-7}
CAPB-10	[No CD] Core damage does not occur (No CD). Therefore, containment integrity is not challenged (No CF).	0.0

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

Table 2-9
FitzPatrick Level 2 Endstates Correlation with the Peach Bottom Unit 2 Collapsed Accident Progression Bins From NUEG/CR-4551

APB	Description	Release Category	Containment End State
1	CD, VB, Early CF, WW, RPV pressure >200 psig at VB	Non-ATWS Early Medium Low, Non-ATWS Early Medium High	Early Wetwell Failure
2	CD, VB, Early CF, WW, RPV pressure <200 psig at VB	Non-ATWS Early Medium Low, Non-ATWS Early Medium High	Early Wetwell Failure
3	CD, VB, Early CF, DW, RPV pressure >200 psig at VB	Early High	Early Drywell Failure
4	CD, VB, Early CF, DW, RPV pressure <200 psig at VB	Early High	Early Drywell Failure
5	CD, VB, Late CF, WW	Late Low, Late Medium Low, Late Medium High	Late Wetwell Failure
6	CD, VB, Late CF, DW	Late High	Late Drywell Failure
7	CD, VB, Vent	ATWS Early Medium Low, ATWS Early Medium High	Bypass
8	CD, VB, No CF	No Containment Failure	No Containment Failure
9	CD, No VB, No CF	No Containment Failure	No Containment Failure
10	No CD	No Containment Failure	No Containment Failure

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

Table 2-10

FitzPatrick 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	2.91×10^{3}
2	Large Containment Isolation Failures (Failure-to-close)	6.79×10^{5}
3a	Small Isolation Failures (Liner breach)	2.91×10^{4}
3b	Large Isolation Failures (Liner Breach)	1.02×10^{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	6.11×10^{5}
7b	Severe Accident Phenomena Induced Early Torus Failures	3.96×10^{5}
7c	Severe Accident Phenomena Induced Late Drywell Failures	5.21×10^{5}
7d	Severe Accident Phenomena Induced Late Torus Failures	3.06×10^{5}
8	Containment Bypassed (ATWS)	4.46×10^{5}

Table 2-11

FitzPatrick 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	2.91×10^{3}	6.19×10^{-7}	1.80×10^{-3}
2	Large Containment Isolation Failures (Failure-to-close)	6.79×10^{5}	2.44×10^{-9}	1.66×10^{-3}
3a	Small Isolation Failures (Liner breach)	2.91×10^{4}	6.59×10^{-8}	1.92×10^{-3}
3b	Large Isolation Failures (Liner Breach)	1.02×10^{5}	6.59×10^{-9}	6.72×10^{-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	6.11×10^{5}	1.74×10^{-7}	1.06×10^{-1}
7b	Severe Accident Phenomena Induced Early Torus Failures	3.96×10^{5}	7.37×10^{-7}	2.92×10^{-1}
7c	Severe Accident Phenomena Induced Late Drywell Failures	5.21×10^{5}	5.82×10^{-8}	3.03×10^{-2}
7d	Severe Accident Phenomena Induced Late Drywell Failures	3.06×10^{5}	4.83×10^{-7}	1.48×10^{-1}
8	Containment Bypassed (ATWS)	4.46×10^{5}	2.97×10^{-7}	1.32×10^{-1}
	Total	3.10×10^{6}	2.44×10^{-6}	7.15×10^{-1}

Table 2-12

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	6.19×10^{-7}	4.49×10^{-7}	3.29×10^{-7}
3a	6.59×10^{-8}	2.20×10^{-7}	3.29×10^{-7}
3b	6.59×10^{-9}	2.20×10^{-8}	3.29×10^{-8}

Table 2-13
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	1.80×10^{-3}	1.31×10^{-3}	9.58×10^{-4}
2	Large Containment Isolation Failures (Failure-to-close)	1.66×10^{-3}	1.66×10^{-3}	1.66×10^{-3}
3a	Small Isolation Failures (Liner breach)	1.92×10^{-3}	6.40×10^{-3}	9.60×10^{-3}
3b	Large Isolation Failures (Liner Breach)	6.72×10^{-4}	2.24×10^{-3}	3.36×10^{-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	1.06×10^{-1}	1.06×10^{-1}	1.06×10^{-1}
7b	Severe Accident Phenomena Induced Early Torus Failures	2.91×10^{-1}	2.91×10^{-1}	2.91×10^{-1}
7c	Severe Accident Phenomena Induced Late Drywell Failures	3.03×10^{-2}	3.03×10^{-2}	3.03×10^{-2}
7d	Severe Accident Phenomena Induced Late Drywell Failures	1.48×10^{-1}	1.48×10^{-1}	1.48×10^{-1}
8	Containment Bypassed (ATWS)	1.32×10^{-1}	1.32×10^{-1}	1.32×10^{-1}
	Total	0.715	0.720	0.724

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 [3,4] 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, 9] extension submittal were used. The assessment also followed previous work as outline in NEI 94-01 [1], the methodology used in EPRI TR-104285 [2], and the NRC Regulatory Guide 1.174 [5].

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 0.36% 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 1.2% 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 1.8% 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 0.56% (0.004 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 1.09 x 10^{-8}/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 2.63 x 10^{-8}/ry. This is also found to be "very small" using the acceptance guidelines in Regulatory Guide 1.174.

7) The change in CCFP of 0.43% 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 FitzPatrick ILRT 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 FitzPatrick 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 0.10% 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 0.34% 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 0.5% 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 0.16% (0.038 person-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 1.03 x 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.

6) The combined internal and external events change in CCFP of 0.20% 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 0.365% 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 1.21% 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 1.82% 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 0.57% (0.0041 person-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 1.22 x 10^{-8}/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 0.60% 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 – Internal Events

	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	6.19×10^{-7}	2.91×10^{3}	1.80×10^{-3}	4.49×10^{-7}	2.91×10^{3}	1.31×10^{-3}	3.29×10^{-7}	2.91×10^{3}	9.58×10^{-4}
2	2.44×10^{-9}	6.79×10^{5}	1.66×10^{-3}	2.44×10^{-9}	6.79×10^{5}	1.66×10^{-3}	2.44×10^{-9}	6.79×10^{5}	1.66×10^{-3}
3a	6.59×10^{-8}	2.91×10^{4}	1.92×10^{-3}	2.20×10^{-7}	2.91×10^{4}	6.40×10^{-3}	3.29×10^{-7}	2.91×10^{4}	9.60×10^{-3}
3b	6.59×10^{-9}	1.02×10^{5}	6.72×10^{-4}	2.20×10^{-8}	1.02×10^{5}	2.24×10^{-3}	3.29×10^{-8}	1.02×10^{5}	3.36×10^{-3}
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	1.74×10^{-7}	6.11×10^{5}	1.06×10^{-1}	1.74×10^{-7}	6.11×10^{5}	1.06×10^{-1}	1.74×10^{-7}	6.11×10^{5}	1.06×10^{-1}
7b	7.37×10^{-7}	3.96×10^{5}	2.91×10^{-1}	7.37×10^{-7}	3.96×10^{5}	2.91×10^{-1}	7.37×10^{-7}	3.96×10^{5}	2.91×10^{-1}
7c	5.82×10^{-8}	5.21×10^{5}	3.04×10^{-2}	5.82×10^{-8}	5.21×10^{5}	3.04×10^{-2}	5.82×10^{-8}	5.21×10^{5}	3.04×10^{-2}
7d	4.83×10^{-7}	3.06×10^{5}	1.48×10^{-1}	4.83×10^{-7}	3.06×10^{5}	1.48×10^{-1}	4.83×10^{-7}	3.06×10^{5}	1.48×10^{-1}
8	2.97×10^{-7}	4.46×10^{5}	1.32×10^{-1}	2.97×10^{-7}	4.46×10^{5}	1.32×10^{-1}	2.97×10^{-7}	4.46×10^{5}	1.32×10^{-1}
Total	2.44×10^{-6}		7.1×10^{-1}	2.44×10^{-6}		7.20×10^{-1}	2.44×10^{-6}		7.24×10^{-1}
ILRT Dose Rate from 3a and 3b			2.59×10^{-3}			8.64×10^{-3}			1.30×10^{-2}
% Of Total			0.36%			1.2%			1.8%
Delta Dose Rate from 3a and 3b (10 to 15 yr)									3.97×10^{-3}
LERF from 3b			6.59×10^{-9}			2.20×10^{-8}			3.29×10^{-8}
Delta LERF (10 to 15 yr)									1.09×10^{-8}
CCFP %			72.0%			72.6%			73.1%
Delta CCFP % (10 to 15 yr)									0.43%

Table 3-2

Summary of Risk Impact on Extending Type A ILRT Test Frequency – Effect of Internal and External Events Risk on FitzPatrick 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	4.28×10^{-6}	2.91×10^{3}	1.25×10^{-2}	2.69×10^{-6}	2.91×10^{3}	7.85×10^{-3}	1.56×10^{-6}	2.91×10^{3}	4.55×10^{-3}
2	5.19×10^{-8}	6.79×10^{5}	3.53×10^{-2}	5.19×10^{-8}	6.79×10^{5}	3.53×10^{-2}	5.19×10^{-8}	6.79×10^{5}	3.53×10^{-2}
3a	6.18×10^{-7}	2.91×10^{4}	1.80×10^{-2}	2.06×10^{-6}	2.91×10^{4}	6.00×10^{-2}	3.09×10^{-6}	2.91×10^{4}	9.00×10^{-2}
3b	6.18×10^{-8}	1.02×10^{5}	6.30×10^{-3}	2.06×10^{-7}	1.02×10^{5}	2.10×10^{-2}	3.09×10^{-7}	1.02×10^{5}	3.15×10^{-2}
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	2.45×10^{-5}	6.11×10^{5}	1.50×10^{1}	2.45×10^{-5}	6.11×10^{5}	1.50×10^{1}	2.45×10^{-5}	6.11×10^{5}	1.50×10^{1}
7b	1.57×10^{-5}	3.96×10^{5}	6.20	1.57×10^{-5}	3.96×10^{5}	6.20	1.57×10^{-5}	3.96×10^{5}	6.20
7c	1.24×10^{-6}	5.21×10^{5}	6.46×10^{-1}	1.24×10^{-6}	5.21×10^{5}	6.46×10^{-1}	1.24×10^{-6}	5.21×10^{5}	6.46×10^{-1}
7d	1.82×10^{-6}	3.06×10^{5}	5.58×10^{-1}	1.82×10^{-6}	3.06×10^{5}	5.58×10^{-1}	1.82×10^{-6}	3.06×10^{5}	5.58×10^{-1}
8	3.65×10^{-6}	4.46×10^{5}	1.63	3.65×10^{-6}	4.46×10^{5}	1.63	3.65×10^{-6}	4.46×10^{5}	1.63
Total	5.19×10^{-5}		2.408×10^{1}	5.19×10^{-5}		2.413×10^{1}	5.19×10^{-5}		2.417×10^{1}
ILRT Dose Rate from 3a and 3b			2.43×10^{-2}			8.10×10^{-2}			1.22×10^{-1}
% Of Total			0.1009%			0.3357%			0.5027%
Delta Dose Rate from 3a and 3b (10 to 15 yr)									**3.80×10^{-2}**
LERF from 3b			6.18×10^{-8}			2.06×10^{-7}			3.09×10^{-7}
Delta LERF (10 to 15 yr)									**1.03×10^{-7}**
CCFP %			90.57%			90.85%			91.05%
Delta CCFP % (10 to 15 yr)									**0.198%**

Table 3-3

Summary of Risk Impact on Extending Type A ILRT Test Frequency – Impact of Containment Steel Liner Corrosion on FitzPatrick 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	6.18×10^{-7}	2.91×10^{3}	1.80×10^{-3}	4.49×10^{-7}	2.91×10^{3}	1.31×10^{-3}	3.26×10^{-7}	2.91×10^{3}	9.52×10^{-4}
2	2.44×10^{-9}	6.79×10^{5}	1.66×10^{-3}	2.44×10^{-9}	6.79×10^{5}	1.66×10^{-3}	2.44×10^{-9}	6.79×10^{5}	1.66×10^{-3}
3a	6.59×10^{-8}	2.91×10^{4}	1.92×10^{-3}	2.20×10^{-7}	2.91×10^{4}	6.40×10^{-3}	3.29×10^{-7}	2.91×10^{4}	9.60×10^{-3}
3b	6.75×10^{-9}	1.02×10^{5}	6.88×10^{-4}	2.29×10^{-8}	1.02×10^{5}	2.34×10^{-3}	3.51×10^{-8}	1.02×10^{5}	3.58×10^{-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	1.74×10^{-7}	6.11×10^{5}	1.06×10^{-1}	1.74×10^{-7}	6.11×10^{5}	1.06×10^{-1}	1.74×10^{-7}	6.11×10^{5}	1.06×10^{-1}
7b	7.37×10^{-7}	3.96×10^{5}	2.91×10^{-1}	7.37×10^{-7}	3.96×10^{5}	2.91×10^{-1}	7.37×10^{-7}	3.96×10^{5}	2.91×10^{-1}
7c	5.82×10^{-8}	5.21×10^{5}	3.04×10^{-2}	5.82×10^{-8}	5.21×10^{5}	3.04×10^{-2}	5.82×10^{-8}	5.21×10^{5}	3.04×10^{-2}
7d	4.83×10^{-7}	3.06×10^{5}	1.48×10^{-1}	4.83×10^{-7}	3.06×10^{5}	1.48×10^{-1}	4.83×10^{-7}	3.06×10^{5}	1.48×10^{-1}
8	2.97×10^{-7}	4.46×10^{5}	1.32×10^{-1}	2.97×10^{-7}	4.46×10^{5}	1.32×10^{-1}	2.97×10^{-7}	4.46×10^{5}	1.32×10^{-1}
Total	2.44×10^{-6}		7.1×10^{-1}	2.44×10^{-6}		7.20×10^{-1}	2.44×10^{-6}		7.24×10^{-1}
ILRT Dose Rate from 3a and 3b			2.61×10^{-3} ($+1.63 \times 10^{-5}$)*			8.74×10^{-3} ($+9.48 \times 10^{-5}$)*			1.32×10^{-2} (+2.21E-04)*
% Of Total			0.365% (+0.0023%)*			1.2128% (+0.0130%)*			1.8199% (+0.0300%)*
Delta Dose Rate from 3a and 3b (10 to 15 yr)									4.09×10^{-3} (+0.0123%)*
LERF from 3b			6.75×10^{-9} ($+1.60 \times 10^{-10}$)*			2.29×10^{-8} ($+9.29 \times 10^{-10}$)*			3.51×10^{-8} ($+2.17 \times 10^{-9}$)*
Delta LERF (10 to 15 yr)									1.22×10^{-8} ($+1.24 \times 10^{-9}$)*
CCFP %			71.99% (+0.0065%)*			72.65% (+0.0381%)*			73.15% (+0.0889%)*
Delta CCFP % (10 to 15 yr)									0.60% (+0.0508%)*

* 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 0.8171%liner 0.0817%floor	Base Case 10%	Base Case 100%	Base Case 1.60×10^{-10}	Base Case 9.29×10^{-10}	Base Case 2.17×10^{-9}	Base Case **1.22×10^{-8}**
Doubles every 2 yrs	Base	Base	Base	4.57×10^{-11}	7.75×10^{-10}	4.50×10^{-9}	**1.47×10^{-8}**
Doubles every 10 yrs	Base	Base	Base	2.37×10^{-10}	3.25×10^{-10}	4.20×10^{-10}	**1.11×10^{-8}**
Base	Base	5%	Base	1.53×10^{-10}	6.20×10^{-11}	1.45×10^{-10}	**1.11×10^{-8}**
Base	Base	15%	Base	1.67×10^{-10}	1.45×10^{-10}	3.38×10^{-10}	**1.12×10^{-8}**
Base	0.166%liner[13] 0.017%floor[13]	Base	Base	3.25×10^{-11}	1.89×10^{-10}	4.42×10^{-10}	**1.12×10^{-8}**
Base	4.07% liner[14] 0.411%floor[14]	Base	Base	7.95×10^{-10}	4.63×10^{-9}	1.08×10^{-8}	**1.72×10^{-8}**
Lower Bound							
Doubles every 10 yrs	Base	5%	10%	4.62×10^{-12}	2.00×10^{-11}	3.66×10^{-11}	**1.10E-08**
Upper Bound							
Doubles every 2 yrs	Base	15%	100%	2.38×10^{-10}	4.03×10^{-9}	2.34×10^{-8}	**3.04E-08**

[13] Base point 10 times lower than base case of 0.0001 at 20 psia.
[14] 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 FitzPatrick 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 1.09×10^{-8}/ry. Since Regulatory Guide 1.174 [6] defines very small changes in LERF as below 10^{-7}/yr, increasing the ILRT interval at FitzPatrick 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 0.56% (0.004 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 $\Delta CCFP_{10\text{-}15}$ is found to be 0.43%. This signifies a very small increase and represents a negligible change in the FitzPatrick 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 internal events, seismic events, and fires events, the $\Delta LERF_{COMBINED10\text{-}15}$ of 1.03×10^{-7}/ry from extending the FitzPatrick 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 2-1), of the acceptance guidelines in NRC Regulatory Guide 1.174 [6]. Therefore, increasing the ILRT interval at FitzPatrick 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 0.16% (0.038 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 0.20%. A change in ΔCCFP 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 1.22×10^{-8}/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 4.1×10^{-3} person-rem/ry or 0.57% 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 0.6%.

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)[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 [2]	2.91×10^{3}	4.49×10^{-7}	1.31×10^{-3}	3.29×10^{-7}	9.58×10^{-4}
2	Containment Isolation System Failure	6.79×10^{5}	2.44×10^{-9}	1.66×10^{-3}	2.44×10^{-9}	1.66×10^{-3}
3a	Small Pre-Existing Failures [2], [3]	2.91×10^{4}	2.20×10^{-7}	6.40×10^{-3}	3.29×10^{-7}	9.60×10^{-3}
3b	Large Pre-Existing Failures [2], [3]	1.02×10^{5}	2.20×10^{-8}	2.24×10^{-3}	3.29×10^{-8}	3.36×10^{-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)[4]	6.11×10^{5}	1.74×10^{-7}	1.06×10^{-1}	1.74×10^{-7}	1.06×10^{-1}
7b	Containment Failure Due to Severe Accident (b) [4]	3.96×10^{5}	7.37×10^{-7}	2.91×10^{-1}	7.37×10^{-7}	2.91×10^{-1}
7c	Containment Failure Due to Severe Accident (c) [4]	5.21×10^{5}	5.82×10^{-8}	3.04×10^{-2}	5.82×10^{-8}	3.04×10^{-2}
7d	Containment Failure Due to Severe Accident (d) [4]	3.06×10^{5}	4.83×10^{-7}	1.48×10^{-1}	4.83×10^{-7}	1.48×10^{-1}
8	Containment Bypass Accidents	4.46×10^{5}	2.97×10^{-7}	1.32×10^{-1}	2.97×10^{-7}	1.32×10^{-1}
TOTALS:			2.44×10^{-6}	0.720	2.44×10^{-6}	0.724
Increase in Dose Rate						0.56%
Increase in LERF					1.09×10^{-8}	
Increase in CCFP (%)					0.43%	

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)[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 [2]	2.91×10^{3}	2.69×10^{-6}	7.85×10^{-3}	1.56×10^{-6}	4.55×10^{-3}
2	Containment Isolation System Failure	6.79×10^{5}	5.19×10^{-8}	3.53×10^{-2}	5.19×10^{-8}	3.53×10^{-2}
3a	Small Pre-Existing Failures [2], [3]	2.91×10^{4}	2.06×10^{-6}	6.00×10^{-2}	3.09×10^{-6}	9.00×10^{-2}
3b	Large Pre-Existing Failures [2], [3]	1.02×10^{5}	2.06×10^{-7}	2.10×10^{-2}	3.09×10^{-7}	3.15×10^{-2}
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)[4]	6.11×10^{5}	2.45×10^{-5}	1.50×10^{1}	2.45×10^{-5}	1.50×10^{1}
7b	Containment Failure Due to Severe Accident (b) [4]	3.96×10^{5}	1.57×10^{-5}	6.20	1.57×10^{-5}	6.20
7c	Containment Failure Due to Severe Accident (c) [4]	5.21×10^{5}	1.24×10^{-6}	6.46×10^{-1}	1.24×10^{-6}	6.46×10^{-1}
7d	Containment Failure Due to Severe Accident (d) [4]	3.06×10^{5}	1.82×10^{-6}	5.58×10^{-1}	1.82×10^{-6}	5.58×10^{-1}
8	Containment Bypass Accidents	4.46×10^{5}	3.65×10^{-6}	1.63	3.65×10^{-6}	1.63
TOTALS:			5.19×10^{-5}	24.13	5.19×10^{-5}	24.17
Increase in Dose Rate						0.16%
Increase in LERF					1.03×10^{-7}	
Increase in CCFP (%)					0.198%	

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

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 (2)	2.91×10^{3}	4.49×10^{-7}	1.31×10^{-3}	3.26×10^{-7}	9.52×10^{-4}
2	Containment Isolation System Failure	6.79×10^{5}	2.44×10^{-9}	1.66×10^{-3}	2.44×10^{-9}	1.66×10^{-3}
3a	Small Pre-Existing Failures (2), (3)	2.91×10^{4}	2.20×10^{-7}	6.40×10^{-3}	3.29×10^{-7}	9.60×10^{-3}
3b	Large Pre-Existing Failures (2), (3)	1.02×10^{5}	2.29×10^{-8}	2.34×10^{-3}	3.51×10^{-8}	3.58×10^{-3}
4	Type B Failures (LLRT)	N/A	0.00E+00	0.00E+00	0.00E+00	0.00E+00
5	Type C Failures (LLRT)	N/A	0.00E+00	0.00E+00	0.00E+00	0.00E+00
6	Other Containment Isolation System Failure	N/A	0.00E+00	0.00E+00	0.00E+00	0.00E+00
7a	Containment Failure Due to Severe Accident (a)(4)	6.11×10^{5}	1.74×10^{-7}	1.06×10^{-1}	1.74×10^{-7}	1.06×10^{-1}
7b	Containment Failure Due to Severe Accident (b) (4)	3.96×10^{5}	7.37×10^{-7}	2.91×10^{-1}	7.37×10^{-7}	2.91×10^{-1}
7c	Containment Failure Due to Severe Accident (c) (4)	5.21×10^{5}	5.82×10^{-8}	3.04×10^{-2}	5.82×10^{-8}	3.04×10^{-2}
7d	Containment Failure Due to Severe Accident (d) (4)	3.06×10^{5}	4.83×10^{-7}	1.48×10^{-1}	4.83×10^{-7}	1.48×10^{-1}
8	Containment Bypass Accidents	4.46×10^{5}	2.97×10^{-7}	1.32×10^{-1}	2.97×10^{-7}	1.32×10^{-1}
TOTALS:			2.44×10^{-6}	0.720	2.44×10^{-6}	0.724
Increase in Dose Rate						0.57%
Increase in LERF					1.22×10^{-8}	
Increase in CCFP (%)					0.60%	

Notes to Tables 15, 16, and 17:

1) The population dose associated with the Technical Specification Leakage is based on scaling the population data, the power level, and allowable Technical Specification leakage compared to the Peach Bottom Unit NUREG/CR-4551 [12] reference plant.

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

3) 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.

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

SECTION 5

REFERENCES

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

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

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

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

(5) U.S. Nuclear Regulatory Commission, "An Approach for Using Probabilistic Risk Assessment In Risk-Informed Decisions On Plant-Specific Changes to the Licensing Basis", Regulatory Guide 1.174, November 2002, Revision 1.

(6) New York Power Authority, "James A. FitzPatrick Nuclear Power Plant Individual Plant Examination," (JAF-RPT-MULTI-02107), April 1998, Revision 1.

(7) U.S. Nuclear Regulatory Commission, "Performance-Based Containment Leak-Test Program", NUREG-1493, September 1995.

(8) Entergy Nuclear Northeast, "Risk Impact Assessment of Extending Containment Type A Test Interval", Calculation Number, IP3-CALC-VC-03357, January 3, 2001 Revision 0.

(9) U. S. Nuclear Regulatory Commission, Indian Point Nuclear Generating Station Unit No. 3 – Issuance of Amendment Re: Frequency of Performance-Based Leakage Rate Testing, April 17, 2001.

(10) Florida Power - Progress Energy, Crystal River Nuclear Plant Letter of June 20, 2001, "Supplemental Risk Informed Information in Support of License Amendment Request No. 267".

(11) U.S. Nuclear Regulatory Commission, "Severe Accident Risks: An Assessment for Five U.S. Nuclear Power Plants, NUREG-1150", December 1990.

(12) Payne et al., "Evaluation of Severe Accident Risks: Peach Bottom, Unit 2," NUREG/CR-4551, SAND86-1309, Volume 4, Rev 1, Part 1, December 1990.

(13) NRC letter to James A. FitzPatrick Nuclear Power Plant issuing Technical Specification Amendment to 234 to implement the requirements of 10 CFR 50, Appendix J, Option B, dated October 4, 1996.

(14) Entergy Nuclear Northeast, "Risk Impact Assessment of Extending Containment Type A Test Interval", Calculation Number, JAF-CALC-PC-04544, June 2, 2003 Revision 0.

(15) KLD Associates, Inc., "Nine Mile Point / James A. FitzPatrick Nuclear Facilities Development of Evacuation Time Estimates," Appendix M "Ingestion Pathway Population," KLD-TR-370, June 2003, Revision 0.

(16) Entergy Nuclear Northeast "James A. FitzPatrick Nuclear Power Plant Updated Final Safety Analysis," Chapter One, Revision 4, May 2001.

(17) NRC letter to James A. FitzPatrick Nuclear Power Plant issuing Technical Specification Amendment to 234 to implement the requirements of 10 CFR 50, Appendix J, Option B, dated October 4, 1996.

(18) Entergy Nuclear Northeast "James A. FitzPatrick Nuclear Power Plant Updated Final Safety Analysis," TABLE 5.2-1, Revision 2, May 2001.

(19) United States Nuclear Regulatory Commission, "Individual Plant Examination of External Events (IPEEE) for Severe Accident Vulnerabilities - 10CFR 50.54(f)," Generic Letter 88-20, Supplement 4, June 28, 1991.

(20) New York Power Authority, "James A. FitzPatrick Nuclear Power Plant Individual Plant Examination for External Events," (JAF-RPT-MISC-02211), June 1996, Revision 0.

(21) 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.

(22) Electric Power Research Institute, "A Methodology for Assessment of Nuclear Power Plant Seismic Margin," EPRI NP-6041-SL, Revision 1, August 1991.

(23) Parkinson, W. J., "EPRI Fire PRA Implementation Guide", prepared by Science Applications International Corporation for Electric Power Research Institute, EPRI TR-105928, December 1995.

(24) 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.

(25) Entergy Nuclear Northeast, "James A. FitzPatrick Nuclear Power Primary Containment Leakage Rate Testing Program Plan," (JAF-RPT-PC-02342), September 2002, Revision 5.

(26) R.P Kennedy, "Overview of Methods for Seismic PRA and Margin Analysis Including Recent Innovations", Proceedings of the OECD-NEA Workshop on Seismic Risk, Tokyo, Japan, August 1999.

(27) Electric Power Research Institute, "Probabilistic Seismic Hazard Evaluation at Nuclear Plant Sites in the Central and Eastern United States: Resolution of the Charleston Issue," EPRI NP-6395-D, April 1989.

(28) Letter from United States Nuclear Regulatory Commission to James Knubel on "James A. FitzPatrick Nuclear Power Plant - Review of FitzPatrick Individual Plant Examination of External Events (IPEEE) SUBMITTAL (TAC NO. M83622).

(29) Entergy Nuclear Northeast, "First Containment Inservice Inspection Interval Program and Schedule," (JAF-RPT-PC-04088), January 2003, Revision 3.

(30) Michael Griesmeyer and L.N. Smith, "A Reference Manual for the Event Progression Analysis Code (EVNTRE)-Section 8," Science Applications International Corporation, NUREG/CR-5174, SAND88-1607, September 1989.

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 FitzPatrick Integrated Leak Rate Testing (ILRT) interval extension risk assessment.

In response to Generic Letter 88-20, Supplement 4 [21], FitzPatrick submitted an Individual Plant Examination of External Events (IPEEE) in June 1996 [20]. 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 FitzPatrick 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 FitzPatrick IPEEE Seismic Analysis

A2.1 Seismic Analysis Methodology Selection

The FitzPatrick 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 FitzPatrick satisfied the NRC requirements for performing a seismic IPEEE as presented in Generic Letter 88-20, Supplement 4 [21]. The methodology comprises an NRC seismic margin assessment (SMA) following the guidance of NUREG-1407 "Procedural and Submittal Guidance for the Individual Plant Examination of External Events (IPEEE) for Severe Accident Vulnerabilities," [21] and EPRI NP-6041 "A Methodology for Assessment of Nuclear Power Plant Seismic Margin," [22] and a containment performance analysis. A seismic margin can be expressed in terms of the earthquake motion level that compromises plant safety--the seismic margin assessment determines whether there is high confidence that the plant can survive a given earthquake. No core damage frequency sequences were quantified as part of the IPEEE seismic risk analysis.

The conclusions of the FitzPatrick IPEEE seismic risk analysis are as follows:

1. The overall plant HCLPF (High Confidence Low Probability of Failure) capacity at JAF is 0.22g PGA.
2. No unique decay heat removal vulnerabilities to seismic events at full power operation were found. Because the overall plant HCLPF capacity with respect to decay heat removal is estimated to be 0.30g PGA, it can be concluded that the decay heat removal pathways are seismically robust with a considerable margin above the 0.15g safe shutdown design basis earthquake.
3. Seismic-induced flooding does not pose major risks.
4. Seismic-induced fires do not pose major risks.
5. No unique seismic induced containment failure mechanisms were identified.

A2.2 Seismic Analysis Assumptions

1) The Simplified Hybrid Method as presented in OECD-NEA Workshop on Seismic Risk, "Overview of Methods for Seismic PRA and Margin Analysis Including Recent Innovations" [26] is used to approximate the FitzPatrick seismic-induced core damage frequency based on the seismic margin analysis results found in the FitzPatrick IPEEE submittal [20].

A2.3 Seismic Analysis Input

1) The FitzPatrick external events individual HCLPF values for each of the seven seismic accident types are as follows [20]:

Seismic Accident Type	Individual HCLPF
Station Blackout	0.22g PGA
Structural	0.30g PGA
Loss-of-Offsite Power	0.27g PGA
Small LOCA	0.27g PGA
Medium LOCA	0.22g PGA
Large LOCA	0.31g PGA
Loss-of-Containment Heat Removal	0.30g PGA

2) The 10% NEP standard normal variable is -1.282. This value is derived from the OECD-NEA Workshop on Seismic Risk [26].

3) The 1% NEP standardized normal variable is -2.326. This value is derived from the OECD-NEA Workshop on Seismic Risk [26].

4) The Simplified Hybrid method presented in OECD-NEA Workshop on Seismic Risk [26] recommends a variable factor β equal to 0.3 to estimate the plant damage seismic risk.

5) The seismic hazard curve for the FitzPatrick site, based upon EPRI NP-6395-D, "Probabilistic Seismic Hazard Evaluation at Nuclear Plant Sites in the Central and Eastern United States: Resolution of the Charleston Issue" [27], is summarized in tabular form in Table A-1.

Table A-1
JAF EPRI Site Seismic Hazard Curve

Acceleration	Frequency of Exceedance (/yr)			
(g)	15%	50%	85%	Mean
0.01	3.50×10^{-3}	9.80×10^{-3}	3.10×10^{-2}	1.50×10^{-2}
0.05	4.70×10^{-5}	2.10×10^{-4}	7.30×10^{-4}	3.60×10^{-4}
0.10	6.60×10^{-6}	5.00×10^{-5}	1.30×10^{-4}	7.30×10^{-5}
0.26	1.90×10^{-7}	4.00×10^{-6}	1.50×10^{-5}	6.90×10^{-6}
0.51	3.10×10^{-9}	3.00×10^{-7}	1.60×10^{-6}	6.90×10^{-7}
0.71	5.30×10^{-10}	5.90×10^{-8}	3.70×10^{-7}	1.80×10^{-7}
1.02	4.00×10^{-10}	7.60×10^{-9}	7.00×10^{-8}	3.60×10^{-8}

A2.4 Seismic Analysis Method of Analysis

Although quantitative risk information is not directly available from the FitzPatrick SMA IPEEE analysis, a paper presented by Robert P. Kennedy [26] provides a method called the Simplified Hybrid Method for obtaining a seismic-induced CDF estimate based on results of an SMA analysis. This methodology requires only the plant HCLPF to estimate the seismic CDF. The approach entails the following steps:

1) Determine the HCLPF seismic capacity C_{HCLPF} from the SMA analysis
2) Estimate the 10% conditional probability of failure capacity $C_{10\%}$
3) Determine hazard exceedance frequency $H_{10\%}$ that corresponds to $C_{10\%}$ from the hazard curve
4) Determine seismic accident type risk P_F

5)` Determine the seismic core damage frequency (SCDF)

Step 1A - Determine the HCLPF seismic capacity C_{HCLPF} from the SMA analysis

The FitzPatrick seismic analysis examined seven seismic accident types; station blackout, structural, loss-of-offsite power, small LOCA, medium LOCA, large LOCA and transient with loss-of-containment heat removal. The respective HCLPF values are:

$C_{HCLPF-SBO}$	=	0.22g	=	station blackout HCLPF	[Section A2.3, Input #1]
$C_{HCLPF-SSC}$	=	0.38g	=	structural components HCLPF	[Section A2.3, Input #1]
$C_{HCLPF-LOSP}$	=	0.27g	=	plant loss-of-offsite HCLPF	[Section A2.3, Input #1]
$C_{HCLPF-S2}$	=	0.27g	=	small LOCA HCLPF	[Section A2.3, Input #1]
$C_{HCLPF-S1}$	=	0.22g	=	medium LOCA HCLPF	[Section A2.3, Input #1]
$C_{HCLPF-A}$	=	0.31g	=	large LOCA HCLPF	[Section A2.3, Input #1]
$C_{HCLPF-TW}$	=	0.30g	=	loss of containment heat removal HCLPF	[Section A2.3, Input #1]

However, the above values do not directly consider the effect of random failures on seismic risk because it uses the HCLPF Maximum/Minimum method to approximate the seismic accident type fragility. Per the OECD-NEA Workshop on Seismic Risk [26] methodology a HCLPF reduction factor is applied to each seismic accident type to account for non-seismic failures and human errors.

Based on a examination of the seismic IPEEE cutsets, a HCLPF reduction factor of 0.7 was selected. As a result, the revised seismic accident type HCLPF values are:

$C_{HCLPF-SBO}$	=	0.22g	/	0.7	=	0.31g (station blackout HCLP)
$C_{HCLPF-SSC}$	=	0.38g	/	0.7	=	0.54g (structural components HCLPF)
$C_{HCLPF-LOSP}$	=	0.27g	/	0.7	=	0.39g (plant loss-of-offsite HCLPF)
$C_{HCLPF-S2}$	=	0.27g	/	0.7	=	0.39g (small LOCA HCLPF)
$C_{HCLPF-S1}$	=	0.22g	/	0.7	=	0.31g (medium LOCA HCLPF)
$C_{HCLPF-A}$	=	0.31g	/	0.7	=	0.44g (large LOCA HCLPF)
$C_{HCLPF-TW}$	=	0.30g	/	0.7	=	0.43g (loss of containment heat removal HCLPF)

Step 2A - Estimate the 10% conditional probability of failure capacity $C_{10\%}$

Per the work presented in OECD-NEA Workshop on Seismic Risk [26], the 10% conditional probability of failure capacity is calculated as follows:

$$C_{10\%} = F_\beta * C_{HCLPF}$$

Where:

$C_{10\%}$ = 10% conditional probability of failure capacity

C_{HCLPF} = seismic accident type HCLPF capacity by the CDFM (Conservative Deterministic Failure Margin) method

$$F_\beta = e^{(NEP10\%-NEP1\%)\beta}$$

Where:

NEP10%	=	10% NEP standard normal variable is	=	-1.282	[Section A2.3, Input #2]
NEP1%	=	1% NEP standard normal variable is	=	-2.326	[Section A2.3, Input #3]
β	=	variable factor	=	0.3	[Section A2.3, Input #4]

Therefore, $F_\beta = e^{[(-1.282)-(-2.326)]*0.3} = 1.37$

Therefore,

$C_{10\%SBO}$ = 1.37 * $C_{HCLPF\text{-}SBO}$ = 1.37 * 0.31 = 0.43g PGA
$C_{10\%SSC}$ = 1.37 * $C_{HCLPF\text{-}SSC}$ = 1.37 * 0.54 = 0.74g PGA
$C_{10\%LOSP}$ = 1.37 * $C_{HCLPF\text{-}LOSP}$ = 1.37 * 0.39 = 0.53g PGA
$C_{10\%S2}$ = 1.37 * $C_{HCLPF\text{-}S2}$ = 1.37 * 0.39 = 0.53g PGA
$C_{10\%S1}$ = 1.37 * $C_{HCLPF\text{-}S1}$ = 1.37 * 0.31 = 0.43g PGA
$C_{10\%A}$ = 1.37 * $C_{HCLPF\text{-}A}$ = 1.37 * 0.44 = 0.61g PGA
$C_{10\%TW}$ = 1.37 * $C_{HCLPF\text{-}TW}$ = 1.37 * 0.43 = 0.59g PGA

Step 3A - Determine hazard exceedance frequency $H_{10\%}$ that corresponds to $C_{10\%}$ from the hazard curve

The seismic hazard curve for the Fitzpatrick site, as presented in Table A-1 is used to determine the hazard exceedance frequency $H_{10\%}$ that corresponds to $C_{10\%}$. These are as follows:

Seismic Accident Type	Acceleration (g)	Mean Frequency of Exceedance (/yr)
$H_{10\%SBO}$	0.31	5.70×10^{-6}
$H_{10\%SSC}$	0.54	6.71×10^{-6}
$H_{10\%LOSP}$	0.39	3.67×10^{-6}
$H_{10\%S2}$	0.39	3.67×10^{-6}
$H_{10\%S1}$	0.31	5.70×10^{-6}
$H_{10\%A}$	0.44	2.43×10^{-6}
$H_{10\%TW}$	0.43	2.68×10^{-6}

Step 4A - Determine seismic accident type risk PF

Per the OECD-NEA Workshop on Seismic Risk [26], the seismic accident type risk is calculated as follows:

$P_{F\text{-}SBO}$ = 0.5 * $H_{10\%SBO}$ = 0.5 * 5.70×10^{-6} = 2.85×10^{-6}/ry
$P_{F\text{-}SSC}$ = 0.5 * $H_{10\%SSC}$ = 0.5 * 6.71×10^{-6} = 3.36×10^{-6}/ry
$P_{F\text{-}LOSP}$ = 0.5 * $H_{10\%LOSP}$ = 0.5 * 3.67×10^{-6} = 1.84×10^{-6}/ry
$P_{F\text{-}S2}$ = 0.5 * $H_{10\%S2}$ = 0.5 * 3.67×10^{-6} = 1.84×10^{-6}/ry
$P_{F\text{-}S1}$ = 0.5 * $H_{10\%S1}$ = 0.5 * 5.70×10^{-6} = 2.85×10^{-6}/ry
$P_{F\text{-}A}$ = 0.5 * $H_{10\%A}$ = 0.5 * 2.43×10^{-6} = 1.21×10^{-6}/ry
$P_{F\text{-}TW}$ = 0.5 * $H_{10\%TW}$ = 0.5 * 2.68×10^{-6} = 1.34×10^{-6}/ry

Step 5A - Determine the seismic core damage frequency (SCDF)

The step involves the summation of the individual seismic accident types frequencies.

$$SCDF = P_{F\text{-}SBO} + P_{F\text{-}SSC} + P_{F\text{-}LOSP} + P_{F\text{-}S1} + P_{F\text{-}S2} + P_{F\text{-}A} + P_{F\text{-}TW}$$

$$SCDF = 2.85 \times 10^{-6} + 3.36 \times 10^{-6} + 1.84 \times 10^{-6} + 1.84 \times 10^{-6} + 2.85 \times 10^{-6} + 1.21 \times 10^{-6} + 1.34 \times 10^{-6}$$

SCDF = 1.53E-05

This information is used in Section A5.0 of this appendix to provide insight into the impact of external hazard risk on the conclusions of this ILRT risk assessment.

A3.0 FitzPatrick IPEEE Fire Analysis

A3.1 Fire Analysis Methodology Selection

The Fire analysis performed for the FitzPatrick IPEEE submittal [20] use the EPRI Fire PRA methodology [23] following the guidance of NUREG-1407 [21]. 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 3.42×10^{-5}/ry [28].

The conclusions of the FitzPatrick IPEEE fire PRA are as follows:

1. The major contribution (61.8 percent) to the fire-induced CDF comes from fires in the cable spreading room, control room, and relay room.

2. The CDF resulting from fires in the cable spreading room may be reduced significantly if the heat detector placement is changed.

3. No significant vulnerabilities were found in an evaluation to resolve unresolved safety issue USI-A45 with respect to decay heat removal fire vulnerabilities.

4. No significant vulnerabilities to water spray, flooding, and CO_2 effects on safe shutdown equipment were found.

5. Except for a specific seismic vulnerability, the CDF induced by hydrogen fires and explosions falls well below the screening criterion of 10^{-6} / year.

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

7. A review of risk issues raised in the Fire Risk Scoping Study [10] concluded that no vulnerabilities exist at FitzPatrick with respect to these issues.

A3.4 Fire Analysis Method of Analysis

The FitzPatrick IPEEE submittal [22, 27] for the fire induced core damage scenarios and the associated frequency results were reviewed in support of this assessment. Based on review of the critical fire areas, the approximate breakdown of the FitzPatrick fire risk profile is as follows:

Fire PRA Accident Type	Frequency (/ry)	% CDF
Cable spreading room	1.13×10^{-5}	0.329
Control room	5.44×10^{-6}	0.159
Relay Room	4.75×10^{-6}	0.139
Reactor building east crescent	1.74×10^{-6}	0.051
Others	1.13×10^{-5}	0.331
Total	3.42×10^{-5}	1.0

This information is used in Section A5.0 of this appendix to provide insight into the impact of external hazard risk on the conclusions of this ILRT risk assessment.

A4.0 Other External Hazards

The FitzPatrick IPEEE submittal [20], 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 Effect of External Hazard Risk on ILRT Risk Assessment Assumptions

1) Because both the seismic margin analysis and fire PRA did not report any seismic-induced accident progression releases, for the purpose of this report, the percent contribution for EPRI accident classes 7b and 7c, are based on the accident class frequencies presented in Table 2-5 (see page 58 of 80) are used. These values are as follows:

   CLASS_7b_% = 7.37×10^{-7} / 2.44×10^{-6} = 0.3019
   CLASS_7c_% = 5.82×10^{-8} / 2.44×10^{-6} = 0.0239

   The remaining ERPI Classes are determined strictly by external events core damage frequencies.

2) Both the seismic margin analysis and the fire PRA are dominated by non-recoverable accident sequences that result in large early releases. Specifically, non-recoverable station blackout accident sequences (seismic margin analysis and fire PRA) and seismic-induced containment failure sequences. From the FitzPatrick IPE submittal [6] station blackout initiated accidents dominate the occurrence of a large early release. This is attributed to drywell failures with no drywell spray operation and either dry or flooded molten core-concrete interaction.

   Per the NEI Guidance Document [4], Enclosure 1, Discussion of Conservatisms in Quantitative Guidance for Delta LERF Impact," specific accident sequences that independently cause a LERF or could never cause a LERF, are to be removed from Class 3b LERF evaluation. Therefore, for the external events impact on the ILRT risk assessment, the evaluation of LERF is performed by multiplying the Class 3b probability by only that portion of core damage frequency that is impacted by Type A ILRT.

A5.2 Effect of External Events Hazard Risk on ILRT Risk Assessment Input

1) Based on the examination in Sections A2.0 through A4.0, the FitzPatrick external event initiated CDF is approximately 3.42×10^{-5}/ry (internal fires) + 1.53×10^{-5}/ry (seismic) = 4.95×10^{-5}/ry.

2) Based on Section A5.2, Assumption#1, the following external event accident sequences are excluded from the Class 3b frequency calculation because they cannot result in a LERF release or independently result in LERF:

- Seismic-induced station blackout sequences, 5.70 x 10^{-6}/ry
- Seismic-induced containment failures sequences, 6.71 x 10^{-6}/ry
- Seismic-induced loss-of-containment heat removal scenarios, 2.68 x 10^{-6}/ry
- Fire-induced station blackout sequences, 2.15 x 10^{-5}/ry (Cable spreading room, Control room, and Relay Room)

A5.3 Effect of External Events Hazard Risk on ILRT Risk Assessment Method of Analysis

The FitzPatrick IPEEE external events risk information presented in Sections A2, A3 and A4 is used to calculate, in accordance with the NEI Interim Guidance [3] for 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.

The change in population dose rate is calculated as outline in Step 7 (section 2.4.7, page 46 of 80) of this report. The results of this calculations when using the information contain in Section A5.2, Assumption #1 and Section A5.3, Input #1, 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	2.91 x 10^{3}	3.66 x 10^{-6}	1.07 x 10^{-2}
2	6.79 x 10^{5}	4.95 x 10^{-8}	3.36 x 10^{-2}
3a	2.91 x 10^{4}	5.52 x 10^{-7}	1.61 x 10^{-2}
3b	1.02 x 10^{5}	5.52 x 10^{-8}	5.63 x 10^{-3}
4	N/A	0.00	0.00
5	N/A	0.00	0.00
6	N/A	0.00	0.00
7a	6.11 x 10^{5}	2.44 x 10^{-5}	1.49 x 10^{1}
7b	3.96 x 10^{5}	1.49 x 10^{-5}	5.91
7c	5.21 x 10^{5}	1.18 x 10^{-6}	6.16 x 10^{-1}
7d	3.06 x 10^{5}	1.34 x 10^{-6}	4.10 x 10^{-1}
8	4.46 x 10^{5}	3.36 x 10^{-6}	1.50
Total		**4.95 x 10^{-5}**	**2.3366 x 10^{1}**

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

EPRI Class	Person-rem	Frequency/Ry	Person-rem/Ry
1	2.91×10^{3}	2.25×10^{-6}	6.55×10^{-3}
2	6.79×10^{5}	4.95×10^{-8}	3.36×10^{-2}
3a	2.91×10^{4}	1.84×10^{-6}	5.36×10^{-2}
3b	1.02×10^{5}	1.84×10^{-7}	1.88×10^{-2}
4	N/A	0.00	0.00
5	N/A	0.00	0.00
6	N/A	0.00	0.00
7a	6.11×10^{5}	2.44×10^{-5}	1.49×10^{1}
7b	3.96×10^{5}	1.49×10^{-5}	5.91
7c	5.21×10^{5}	1.18×10^{-6}	6.16×10^{-1}
7d	3.06×10^{5}	1.34×10^{-6}	4.10×10^{-1}
8	4.46×10^{5}	3.36×10^{-6}	1.50
Total		**4.95×10^{-5}**	**2.3412×10^{1}**

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

EPRI Class	Person-rem	Frequency/Ry	Person-rem/Ry
1	2.91×10^{3}	1.23×10^{-6}	3.60×10^{-3}
2	6.79×10^{5}	4.95×10^{-8}	3.36×10^{-2}
3a	2.91×10^{4}	2.76×10^{-6}	8.04×10^{-2}
3b	1.02×10^{5}	2.76×10^{-7}	2.81×10^{-2}
4	N/A	0.00	0.00
5	N/A	0.00	0.00
6	N/A	0.00	0.00
7a	6.11×10^{5}	2.44×10^{-5}	1.49×10^{1}
7b	3.96×10^{5}	1.49×10^{-5}	5.91
7c	5.21×10^{5}	1.18×10^{-6}	6.16×10^{-1}
7d	3.06×10^{5}	1.34×10^{-6}	4.10×10^{-1}
8	4.46×10^{5}	3.36×10^{-6}	1.50
Total		**4.95×10^{-5}**	**2.3446×10^{1}**

Based on the results summarized above and those presented in Table 2-13 (see page 64 of 80), for the current FitzPatrick 1-in10 years ILRT interval, the percentage contribution to total dose rate from EPRI's accident Classes 3a and 3b is calculated as follows:

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

Where:

$PER\_CHG_{COMBINED-10}$ = 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

$CLASS\_3a\_DOSE_{COMBINED-10}$ = 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 = 6.40×10^{-3} /ry [Table 2-13]

$CLASS\_3b\_DOSE_{INTERNAL-10}$ = internal events EPRI accident Class 3b dose rate given a 1-in-10 years ILRT interval = 2.24×10^{-3} /ry [Table 2-13]

$CLASS\_3a\_DOSE_{EXTERNAL-10}$ = external events EPRI accident Class 3a dose rate given a 1-in-10 years ILRT interval = 5.36×10^{-2} person-rem/ry [See 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 = 1.88×10^{-2} person-rem/ry [See 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 = 0.720 (person-rem/ry) [Table 2-13]

$TOT\text{-}DOSE_{EXTERNAL-10}$ = Total external events dose rate for all EPRI's Classes given a 1-in-10 years ILRT interval = 23.412 (person-rem/ry)[See 1-in-10 years table above]

Therefore,

$$PER\_CHG_{COMBINED-10} = \left[\frac{(6.40 \times 10^{-3} + 5.36 \times 10^{-2}) + (2.24 \times 10^{-3} + 1.88 \times 10^{-2})}{0.720 + 23.412}\right] * 100$$

$$PER\_CHG_{COMBINED-10} = 0.34\%$$

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:

$PER\_CHG_{COMBINED-15}$ = 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

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

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

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

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

$CLASS\_3a\_DOSE_{INTERNAL-15}$ = internal events EPRI accident Class 3a dose rate given a 1-in-15 years ILRT interval = 9.60×10^{-3} person-rem/ry [Table 2-13]

$CLASS\_3b\_DOSE_{INTERNAL-15}$ = internal events EPRI accident Class 3b dose rate given a 1-in-15 years ILRT interval = 3.36×10^{-3} person-rem/ry [Table 2-13]

$CLASS\_3a\_DOSE_{EXTERNAL-15}$ = external events EPRI accident Class 3a dose rate given a 1-in-15 years ILRT interval = 8.04×10^{-2} person-rem/ry [See 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 = 2.81×10^{-2} person-rem/ry [See 1-in-15 years table above]

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

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

$TOT\text{-}DOSE_{INTERNAL-15}$ = Total internal events dose rate for all EPRI's Classes given a 1-in-15 years ILRT interval = 0.724 (person-rem/ry) [Table 2-13]

$TOT\text{-}DOSE_{EXTERNAL-15}$ = Total external events dose rate for all EPRI's Classes given a 1-in-15 years ILRT interval = 23.446 (person-rem/ry)[See 1-in-10 years table above]

Therefore,

$$PER\_CHG_{COMBINED-15} = \left[\frac{(9.60 \times 10^{-3} + 8.04 \times 10^{-2}) + (3.36 \times 10^{-3} + 2.81 \times 10^{-2})}{0.724 + 23.446}\right] * 100$$

$PER\_CHG_{COMBINED-15}$ = 0.50%

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 = 0.724 (person-rem/ry) [Table 2-13]

$TOT\text{-}DOSE_{EXTERNAL\text{-}15}$ = Total external events dose rate for all EPRI's Classes given a 1-in-10 years ILRT interval = 23.446 (person-rem/ry)[See 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 = 0.720 (person-rem/ry) [Table 2-13]

$TOT\text{-}DOSE_{EXTERNAL\text{-}10}$ = Total external events dose rate for all EPRI's Classes given a 1-in-10 years ILRT interval = 23.412 (person-rem/ry) [See 1-in-10 years table above]

Therefore,

$$INCREASE_{COMBINED10\text{-}15} = \left[\frac{(23.446 + 0.724) - (23.412 + 0.720)}{(23.412 + 0.720)}\right] * 100$$

$$INCREASE_{COMBINED10\text{-}15} = 0.16\%$$

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 0.16%. This value can be considered to be a negligible increase in risk.

<u>Evaluate the External Events Hazard Risk Impact In Terms of LERF</u>

This step, per the NEI Interim Guidance [3] 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:

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

Where:

$\Delta LERF_{COMBINED10\text{-}15}$ = 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 = 3.29×10^{-8}/ry [Table 2-12]

$CLASS\_3b_{EXTERNAL\text{-}15}$ = external events frequency of EPRI accident Class 3b given a 1-in-15 years ILRT Interval = 2.76×10^{-7}/ry [See 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 = 2.20×10^{-8}/ry [Table 2-12]

$CLASS\_3b_{EXTERNAL\text{-}10}$ = external events frequency of EPRI accident Class 3b given a 1-in-10 years ILRT Interval = 1.84×10^{-7}/ry [See 1-in-10 years table above]

Therefore,

$$\Delta LERF_{COMBINED10\text{-}15} = (3.29 \times 10^{-8} + 2.76 \times 10^{-7}) - (2.20 \times 10^{-8} + 1.84 \times 10^{-7})$$

$$\Delta LERF_{COMBINED10\text{-}15} = 1.03 \times 10^{-7}/ry$$

The risk acceptance criteria of Regulatory Guide 1.174 as previously discussed in Section 2.4.8, Step 8 of this report, 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" [5], provides NRC recommendations for using risk information in support of applications requesting changes to the license basis of the plant.

The $\Delta LERF_{COMBINED10\text{-}15}$ of 1.03×10^{-7}/ry from extending the FitzPatrick LRT 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 2-1), of the acceptance guidelines in NRC Regulatory Guide 1.174 [5]. Therefore, increasing the ILRT interval at FitzPatrick from the currently allowed 1-in-10 years to 1-in-15 years is non-risk significant from a risk perspective.

Evaluate the External Events Hazard Change In Conditional Containment Failure Probability

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

Similar to Step 9 (Section 2.4.9) of this report, 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-([Class 1 frequency + Class 3a frequency]/CDF)}*100, %

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

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

Where:

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

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

= $CLASS\_1_{INTERNAL-10}$ + $CLASS\_1_{EXTERNAL-10}$

$CLASS\_1_{INTERNAL-10}$ = internal events frequency of EPRI accident Class 1given a 1-in-10 years ILRT interval = 4.49×10^{-7}/ry [Table 2-12]

$CLASS\_1_{EXTERNAL-10}$ = external events frequency of EPRI accident Class 1given a 1-in-10 years ILRT interval = 2.25×10^{-6}/ry [See 1-in-10 years table above]

$CLASS\_3a_{INTERNAL-10}$ = internal events frequency of EPRI accident Class 3a given a 1-in-10 years ILRT interval = 2.20×10^{-7}/ry [See 1-in-10 years table above]

$CLASS\_3a_{EXTERNAL-10}$ = external events frequency of EPRI accident Class 3a given a 1-in-10 years ILRT interval = 1.84×10^{-6}/ry /ry [See 1-in-10 years table above]

$CDF_{COMBINED}$ = FitzPatrick combined internal events and external events CDF
= 2.44×10^{-6}/ry [Section 5, input#2] + 4.95×10^{-5}/ry [Section A5.3, input#1]
= 5.19×10^{-5}/ry

Therefore,

$$CCFP_{COMBINED-10} = \left\{ 1 - \left(\left[\frac{(4.49 \times 10^{-7} + 2.25 \times 10^{-6}) + (1.84 \times 10^{-6} + 2.20 \times 10^{-7})}{5.19 \times 10^{-5}} \right] \right) \right\} * 100\%$$

$CCFP_{COMBINED-10}$ = 90.8%

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

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

Where:

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

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

= $CLASS\_1_{INTERNAL-15}$ + $CLASS\_1_{EXTERNAL-15}$

$CLASS\_1_{INTERNAL-15}$ = internal events frequency of EPRI accident Class 1given a 1-in-15 years ILRT interval = 3.29×10^{-7}/ry [Table 2-12]

$CLASS\_1_{EXTERNAL-15}$ = external events frequency of EPRI accident Class 1given a 1-in-15 years ILRT interval = 1.23×10^{-6}/ry [See 1-in-15 years table above]

$CLASS\_3a_{INTERNAL-15}$ = internal events frequency of EPRI accident Class 3a given a 1-in-15 years ILRT interval = 3.29×10^{-7}/ry [See 1-in-15 years table above]

$CLASS\_3a_{EXTERNAL-15}$ = external events frequency of EPRI accident Class 3a given a 1-in-15 years ILRT interval = 2.76×10^{-6}/ry /ry [See 1-in-15 years table above]

$CDF_{COMBINED}$ = FitzPatrick combined internal events and external events CDF
= 2.44×10^{-6}/ry [Section 5, input#2] + 4.95×10^{-5}/ry [Section A5.3, input#1]
= 5.19×10^{-5}/ry

Therefore,

$$CCFP_{COMBINED-15} = \left\{ 1 - \left(\left[\frac{(3.29 \times 10^{-7} + 1.23 \times 10^{-6}) + (3.29 \times 10^{-7} + 2.76 \times 10^{-6})}{5.19 \times 10^{-5}} \right] \right) \right\} * 100\%$$

$CCFP_{COMBINED-15}$ = 91.0%

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:

$\Delta CCFP_{COMBINED10-15}$ = $CCFP_{COMBINED15}$ - $CCFP_{COMBINED10}$

$\Delta CCFP_{COMBINED10-15}$ = 91.0% - 90.8%

$\Delta CCFP_{COMBINED10-15}$ = 0.20%

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-1. The combined internal and external events effect on the ILRT risk is shown in Table A-3. This Table combines the results of Table 11, 12, and 13 with the results depicted in Table A-2.

A6.0 Conclusions

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

1. The $\Delta LERF_{COMBINED10-15}$ of 1.03 x 10^{-7}/ry from extending the FitzPatrick 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 2-1), of the acceptance guidelines in NRC Regulatory Guide 1.174 [5]. Therefore, increasing the ILRT interval at FitzPatrick 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 0.16% (0.038 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 0.20%. A change in CCFP of less than 1% is insignificant from a risk perspective.

Table A-2

Effect of External Events Hazard Risk on FitzPatrick 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	3.66×10^{-6}	2.91×10^{3}	1.07×10^{-2}	2.25×10^{-6}	2.91×10^{3}	6.55×10^{-3}	1.23×10^{-6}	2.91×10^{3}	3.60×10^{-3}
2	4.95×10^{-8}	6.79×10^{5}	3.36×10^{-2}	4.95×10^{-8}	6.79×10^{5}	3.36×10^{-2}	4.95×10^{-8}	6.79×10^{5}	3.36×10^{-2}
3a	5.52×10^{-7}	2.91×10^{4}	1.61×10^{-2}	1.84×10^{-6}	2.91×10^{4}	5.36×10^{-2}	2.76×10^{-6}	2.91×10^{4}	8.04×10^{-2}
3b	5.52×10^{-8}	1.02×10^{5}	5.63×10^{-3}	1.84×10^{-7}	1.02×10^{5}	1.88×10^{-2}	2.76×10^{-7}	1.02×10^{5}	2.81×10^{-2}
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	2.44×10^{-5}	6.11×10^{5}	1.49×10^{1}	2.44×10^{-5}	6.11×10^{5}	1.49×10^{1}	2.44×10^{-5}	6.11×10^{5}	1.49×10^{1}
7b	1.49×10^{-5}	3.96×10^{5}	5.91	1.49×10^{-5}	3.96×10^{5}	5.91	1.49×10^{-5}	3.96×10^{5}	5.91
7c	1.18×10^{-6}	5.21×10^{5}	6.16×10^{-1}	1.18×10^{-6}	5.21×10^{5}	6.16×10^{-1}	1.18×10^{-6}	5.21×10^{5}	6.16×10^{-1}
7d	1.34×10^{-6}	3.06×10^{5}	4.10×10^{-1}	1.34×10^{-6}	3.06×10^{5}	4.10×10^{-1}	1.34×10^{-6}	3.06×10^{5}	4.10×10^{-1}
8	3.36×10^{-6}	4.46×10^{5}	1.50	3.36×10^{-6}	4.46×10^{5}	1.50	3.36×10^{-6}	4.46×10^{5}	1.50
Total	**4.95×10^{-5}**		**2.337×10^{1}**	**4.95×10^{-5}**		**2.341×10^{1}**	**4.95×10^{-5}**		**2.345×10^{1}**
ILRT Dose Rate from 3a and 3b			2.17×10^{-2}			7.24×10^{-2}			1.09×10^{-1}
% Of Total			0.0929%			0.3091%			0.4630%
Delta Dose Rate from 3a and 3b (10 to 15 yr)									**3.32×10^{-2}**
LERF from 3b			5.52×10^{-8}			1.84×10^{-7}			2.76×10^{-7}
Delta LERF (10 to 15 yr)									**9.20×10^{-8}**
CCFP %			91.48%			91.74%			91.93%
Delta CCFP % (10 to 15 yr)									**0.186%**

Table A-3

Effect of Internal and External Events Risk on FitzPatrick 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	4.28×10^{-6}	2.91×10^{3}	1.25×10^{-2}	2.69×10^{-6}	2.91×10^{3}	7.85×10^{-3}	1.56×10^{-6}	2.91×10^{3}	4.55×10^{-3}
2	5.19×10^{-8}	6.79×10^{5}	3.53×10^{-2}	5.19×10^{-8}	6.79×10^{5}	3.53×10^{-2}	5.19×10^{-8}	6.79×10^{5}	3.53×10^{-2}
3a	6.18×10^{-7}	2.91×10^{4}	1.80×10^{-2}	2.06×10^{-6}	2.91×10^{4}	6.00×10^{-2}	3.09×10^{-6}	2.91×10^{4}	9.00×10^{-2}
3b	6.18×10^{-8}	1.02×10^{5}	6.30×10^{-3}	2.06×10^{-7}	1.02×10^{5}	2.10×10^{-2}	3.09×10^{-7}	1.02×10^{5}	3.15×10^{-2}
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	2.45×10^{-5}	6.11×10^{5}	1.50×10^{1}	2.45×10^{-5}	6.11×10^{5}	1.50×10^{1}	2.45×10^{-5}	6.11×10^{5}	1.50×10^{1}
7b	1.57×10^{-5}	3.96×10^{5}	6.20	1.57×10^{-5}	3.96×10^{5}	6.20	1.57×10^{-5}	3.96×10^{5}	6.20
7c	1.24×10^{-6}	5.21×10^{5}	6.46×10^{-1}	1.24×10^{-6}	5.21×10^{5}	6.46×10^{-1}	1.24×10^{-6}	5.21×10^{5}	6.46×10^{-1}
7d	1.82×10^{-6}	3.06×10^{5}	5.58×10^{-1}	1.82×10^{-6}	3.06×10^{5}	5.58×10^{-1}	1.82×10^{-6}	3.06×10^{5}	5.58×10^{-1}
8	3.65×10^{-6}	4.46×10^{5}	1.63	3.65×10^{-6}	4.46×10^{5}	1.63	3.65×10^{-6}	4.46×10^{5}	1.63
Total	**5.19×10^{-5}**		**2.408×10^{1}**	**5.19×10^{-5}**		**2.413×10^{1}**	**5.19×10^{-5}**		**2.417×10^{1}**
ILRT Dose Rate from 3a and 3b			2.43×10^{-2}			8.10×10^{-2}			1.22×10^{-1}
% Of Total			0.1009%			0.3357%			0.5027%
Delta Dose Rate from 3a and 3b (10 to 15 yr)									**3.80×10^{-2}**
LERF from 3b			6.18×10^{-8}			2.06×10^{-7}			3.09×10^{-7}
Delta LERF (10 to 15 yr)									**1.03×10^{-7}**
CCFP %			90.57%			90.85%			91.05%
Delta CCFP % (10 to 15 yr)									0.198%

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 [24] 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 determines the total likelihood of non-detected containment leakage given a change in the likelihood given 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 [24], 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 evaluated 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 [24], a half failure is assumed for basemat concealed liner corrosion due to the lack of identified failures.

2) Consistent with the Calvert Cliffs methodology [24], 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 [24], 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 [24], 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 [24], 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 [24], 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 [24], 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 [24], 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 inspect ion.

2) The number of steel-lined containments is 70 [24].

3) The exposure time in detecting a containment flaw is 5.5 years. This is consistent with the Calvert Cliffs methodology [24] 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 [24], 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 155psia psia [6].

B5.0 Steel Shell Corrosion Analysis

<u>Step 1B - Determine the Historical Liner Flaw Likelihood.</u>

This step calculates historical liner flaw likelihood consistent wit the Calvert Cliffs mythology [24]. This value, for FitzPatrick'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:

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

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

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

The inaccessible area of the drywell floor

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

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 [24], 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 [24]. The results of Step 2B are use to generate these values as follows:

Accessible portion of the drywell and torus,

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

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

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

Inaccessible portion of the drywell and submergence area of the torus,

$$IDTFLAW_{3\text{-}10} = \sum_{i=1,3} IDTF_{RATEi}$$

$$IDTFLAW_{1\text{-}10} = \sum_{i=1,10} IDTF_{RATEi}$$

$$IDTFLAW_{1\text{-}15} = \sum_{i=1,15} IDTF_{RATEi}$$

[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

$$DFFLAW_{3\text{-}10} = \sum_{i=1,3} DFF_{RATEi}$$

$$DFFLAW_{1\text{-}10} = \sum_{i=1,10} DFF_{RATEi}$$

$$DFFLAW_{1\text{-}15} = \sum_{i=1,15} DFF_{RATEi}$$

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

$ADTF_{RATEi}$ = aged adjusted liner flaw likelihood, given accessible portion of the drywell and torus (Table B-1)

$IDTF_{RATEi}$ = aged adjusted liner flaw likelihood, given inaccessible portion of the drywell and submergence area of the torus (Table B-1)

DFF_{RATEi} = 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 [24] with a FitzPatrick specific value for the upper-end pressure failure (100% likelihood) taken from Section 4.5 of the IPE [7]. A containment pressure of 155 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 [24]. Per the Calvert Cliffs methodology [24], 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 155 psia provide sufficient information to solve for the slope m, and the intercept b, as follows:

Slope m,

$$m = LN(FP(100\%) - LN(0.1\%) \ / \ (\text{Upper Pressure} - \text{Lower Pressure})$$

$$m = LN(1.0) - LN(0.001) \ / \ (155\text{-}20)$$

$$m = 5.12 \times 10^{-2}$$

Intercept b,

$$b = FP(100\%) \ / \ e^{m*p}$$

$$b = 1 \ / \ e^{5.12 \times 10\text{-}2*155}$$

$$b = 3.56 \times 10^{-4}$$

The FitzPatrick March 7, 1995 ILRT used a test pressure of 46.5 psig (or 61.2 psia) [25]. Based on this pressure the likelihood of containment breach in the liner is:

$$FP\ (61.2\ psia) = 3.56 \times 10^{-4} * e^{5.12 \times 10^{-2} * 61.2}$$

$$FP\ (61.2\ psia) = 0.0082 \quad \text{or} \quad 0.82\%$$

For the Drywell floor, the failure probability is set to one-tenth of the failure probability for Drywell walls, or 0.082%. (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 FitzPatrick. 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 (100 percent internal and 75 percent external) [29] and torus (100 percent external and 100 percent of the area above the water line) [29] is set to 10%, consistent with the Calvert Cliffs analysis [24]. 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 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 [24], 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 (61.2 psia) given liner flaw and accessible portion of the drywell and torus = 0.0082 (0.82%) [Step 4B]

ADTVISUAL = visual inspection detection failure accessible portion of the drywell and torus = 0.1 (10%) [Step 5B]

Therefore,

$$ADTLEAK_{3\text{-}10} = 0.0071 * 0.0082 * 0.1 = 5.822 \times 10^{-6} \ (0.0005822\%)$$

$$ADTLEAK_{1\text{-}10} = 0.0414 * 0.0082 * 0.1 = 3.395 \times 10^{-5} \ (0.0033948\%)$$

$$ADTLEAK_{1\text{-}15} = 0.0968 * 0.0082 * 0.1 = 7.938 \times 10^{-5} \ (0.0079376\%)$$

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]

$IDTFLAW_{1-10}$ = 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]

$IDTFLAW_{1-15}$ = 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 (61.2 psia) given liner flaw and inaccessible portion of the drywell and submergence area of the torus
= 0.0082 (0.82%) [Step 4B]

IDTVISUAL = visual inspection detection failure inaccessible portion of the drywell and submergence area of the torus = 1.0 (100%) [Step 5B]

Therefore,

$IDTLEAK_{3-10}$ = 0.0071 * 0.0082 * 1.0 = 5.822×10^{-5} (0.005822%)

$IDTLEAK_{1-10}$ = 0.0414 * 0.0082 * 1.0 = 3.395×10^{-4} (0.033948%)

$IDTLEAK_{1-15}$ = 0.0968 * 0.0082 * 1.0 = 7.938×10^{-4} (0.079376%)

Inaccessible portion of the drywell floor,

$DFLEAK_{3-10}$ = $DFTFLAW_{3-10}$ * $DFTFP_{ILRT}$ * DFTVISUAL

$DFTLEAK_{1-10}$ = $DFTFLAW_{1-10}$ * $DFTFP_{ILRT}$ * DFTVISUAL

$DFTLEAK_{1-15}$ = $DFTFLAW_{1-15}$ * $DFTFP_{ILRT}$ * DFTVISUAL

Where:

$DFLEAK_{3-10}$ = likelihood of non-detected containment leakage, given 3-in-10 years test interval and inaccessible portion of the drywell floor

$DFLEAK_{1-10}$ = likelihood of non-detected containment leakage, given 1-in-10 years test interval and inaccessible portion of the drywell floor

$DFLEAK_{1-15}$ = likelihood of non-detected containment leakage, given 1-in-15 years test interval and inaccessible portion of the drywell floor

$DFFLAW_{3-10}$ = 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]

$DFFLAW_{1-10}$ = 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]

$DFFLAW_{1-15}$ = 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 (61.2 psia) given liner flaw and inaccessible portion of the drywell floor = 0.00082 (0.082%) [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 * 0.00082 * 1.0 = 1.476×10^{-6} (0.0001476%)

$DFTLEAK_{1\text{-}10}$ = 0.0104 * 0.00082 * 1.0 = 8.528×10^{-6} (0.000853%)

$DFTLEAK_{1\text{-}15}$ = 0.0242 * 0.00082 * 1.0 = 1.984×10^{-5} (0.001984%)

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,

$TOTAL_{3\text{-}10}$ = 0.00058% + 0.00582% + 0.00016% = 0.00655%

$TOTAL_{1\text{-}10}$ = 0.00340% + 0.03390% + 0.00085% = 0.0385%

$TOTAL_{1\text{-}15}$ = 0.00794% + 0.07938% + 0.00198% = 0.08975%

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 Step 7 (Section 2.4.7, page 46 of 80), of this report. The results of this calculations, is presented below as follows:

For 3-in-10 years,

EPRI Class	Person-rem	Frequency/Ry		Person-rem/Ry
1	2.91×10^{3}	6.18×10^{-7}		1.80×10^{-3}
2	6.79×10^{5}	2.44×10^{-9}	Corrosion Addition	1.66×10^{-3}
3a	2.91×10^{4}	6.59×10^{-8}		1.92×10^{-3}
3b	1.02×10^{5}	6.75×10^{-9}	1.60×10^{-10}	6.88×10^{-4}
4	N/A	0.0		0.0
5	N/A	0.0		0.0
6	N/A	0.0		0.0
7a	6.11×10^{5}	1.74×10^{-7}		1.06×10^{-1}
7b	3.96×10^{5}	7.37×10^{-7}		2.91×10^{-1}
7c	5.21×10^{5}	5.82×10^{-8}		3.04×10^{-2}
7d	3.06×10^{5}	4.83×10^{-7}		1.48×10^{-1}
8	4.46×10^{5}	2.97×10^{-7}		1.32×10^{-1}
Total		2.44×10^{-6}		0.7147

ILRT Dose Rate from 3a and 3b = 1.92×10^{-3} + 6.88 x 10-4 = 2.61×10^{-3} person-rem/ry
% Of Total = 100 * [1.92×10^{-3} + 6.88×10^{-4}] / 0.7147 = 0.3650%
LERF from 3b = 6.75×10^{-9} /ry
$CCFP\%_{LINER3\text{-}10}$ = 1 - [6.18×10^{-7} + 6.59×10^{-8}] / 2.44×10^{-6} = 72.0%

For 1-in-10 years,

EPRI Class	Person-rem	Frequency/Ry		Person-rem/Ry
1	2.91×10^{3}	4.49×10^{-7}		1.31×10^{-3}
2	6.79×10^{5}	2.44×10^{-9}	Corrosion Addition	1.66×10^{-3}
3a	2.91×10^{4}	2.20×10^{-7}		6.40×10^{-3}
3b	1.02×10^{5}	2.29×10^{-8}	9.29×10^{-10}	2.34×10^{-3}
4	N/A	0.0		0.0
5	N/A	0.0		0.0
6	N/A	0.0		0.0
7a	6.11×10^{5}	1.74×10^{-7}		1.06×10^{-1}
7b	3.96×10^{5}	7.37×10^{-7}		2.91×10^{-1}
7c	5.21×10^{5}	5.82×10^{-8}		3.04×10^{-2}
7d	3.06×10^{5}	4.83×10^{-7}		1.48×10^{-1}
8	4.46×10^{5}	2.97×10^{-7}		1.32×10^{-1}
Total		**2.44×10^{-6}**		**0.7203**

ILRT Dose Rate from 3a and 3b = 6.40×10^{-3} + 2.34×10^{-3} = 8.74×10^{-3} person-rem/ry
% Of Total = 100 * [6.4×10^{-3} + 2.34×10^{-3}] / 0.7203 = 1.213%
LERF from 3b = 2.29×10^{-8} /ry
$CCFP\%_{LINER1\text{-}10}$ = 1 - [4.49×10^{-7} + 2.20×10^{-7}] / 2.44×10^{-6} = 72.6%

For 1-in-15 years,

EPRI Class	Person-rem	Frequency/Ry		Person-rem/Ry
1	2.91×10^{3}	3.26×10^{-7}		9.52×10^{-3}
2	6.79×10^{5}	2.44×10^{-9}	Corrosion Addition	1.66×10^{-3}
3a	2.91×10^{4}	3.29×10^{-7}		9.60×10^{-3}
3b	1.02×10^{5}	3.51×10^{-8}	2.17×10^{-9}	3.58×10^{-3}
4	N/A	0.0		0.0
5	N/A	0.0		0.0
6	N/A	0.0		0.0
7a	6.11×10^{5}	1.74×10^{-7}		1.06×10^{-1}
7b	3.96×10^{5}	7.37×10^{-7}		2.91×10^{-1}
7c	5.21×10^{5}	5.82×10^{-8}		3.04×10^{-2}
7d	3.06×10^{5}	4.83×10^{-7}		1.48×10^{-1}
8	4.46×10^{5}	2.97×10^{-7}		1.32×10^{-1}
Total		**2.44×10^{-6}**		**0.7244**

ILRT Dose Rate from 3a and 3b = 9.60×10^{-3} + 3.58×10^{-3} = 1.32×10^{-2} person-rem/ry
% Of Total = 100 * [9.60×10^{-3} + 3.58×10^{-3}] / 0.7244 = 1.819%
LERF from 3b = 3.51×10^{-8} /ry
$CCFP\%_{LINER1\text{-}15}$ = 1 - [3.26×10^{-7} + 3.29×10^{-7}] / 2.44×10^{-6} = 73.2%

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
= 0.7244 (person-rem/ry) [See 1-in-15 years table above]

$TOT\text{-}DOSE_{RATE\text{-}LINER15}$ = Total dose rate for all EPRI's Classes given a 1-in-10 years ILRT interval
= 0.7203 (person-rem/ry) [See 1-in-10 years table above]

Therefore,

$$INCREASE_{LINER10\text{-}15} = \left[\frac{0.7244 - 0.7203}{0.7203}\right] * 100 = 0.57\%$$

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 0.57%. 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:

$$\Delta LERF_{LNER10\text{-}15} = CLASS\_3b\_FREQUENC\ Y_{LNER15} - CLASS\_3b\_FREQUENC\ Y_{LNER10}$$

Where:

$\Delta LERF_{LNER10\text{-}15}$ = the change in LERF from 1-in-10 years ILRT interval to 1-in-15 years ILRT interval

$CLASS\_3b\_FREQUENC\ Y_{LNER15}$ = frequency of EPRI accident Class 3b given a 1-in-15 years ILRT interval = 3.51×10^{-8} /ry [Step 7B]

$CLASS\_3b\_FREQUENC\ Y_{LNER10}$ = frequency of EPRI accident Class 3b given a 1-in-10 years ILRT interval = 2.29×10^{-8}/ry [Step 7B]

Therefore,

$$\Delta LERF_{LNER10\text{-}15} = 3.51 \times 10^{-8} - 2.29 \times 10^{-8}$$

$$\Delta LERF_{LNER10\text{-}15} = 1.22 \times 10^{-8}/ry$$

Based on this result, the inclusion of corrosion effects in the ILRT assessment would not change the previous conclusions of this report (See Section 2.4). That is, the change in LERF from extending the interval to 15 years from the current 10 years requirement is estimated to be about 1.22 x 10^{-8}/ry. This value is below the NRC Regulatory Guide 1.174 [5] of 10^{-7}/yr. Therefore, because Regulatory Guide 1.174 [5] defines very small changes in LERF as below 10^{-7}/yr, increasing the ILRT interval at FitzPatrick 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:

$$\Delta LERF_{LNER3\text{-}15} = CLASS\_3b\_FREQUENCY_{LNER15} - CLASS\_3b\_FREQUENCY_{LNER3}$$

Where:

$\Delta LERF_{LNER3\text{-}15}$ = the change in LERF from 3-in-10 years ILRT interval to 1-in-15 years ILRT interval

$CLASS\_3b\_FREQUENCY_{LNER15}$ = frequency of EPRI accident Class 3b given a 1-in-15 years ILRT interval = 3.51 x 10^{-8} /ry [Step 7B]

$CLASS\_3b\_FREQUENCY_{LNER3}$ = frequency of EPRI accident Class 3b given a 1-in-10 years ILRT interval = 6.75 x 10^{-9}/ry [Step 7B]

Therefore,

$$\Delta LERF_{LNER3\text{-}15} = 3.51 \times 10^{-8} - 6.75 \times 10^{-9}$$

$$\Delta LERF_{LNER3\text{-}15} = 2.84 \times 10^{-8}/ry$$

Step 9B - Evaluate the change in conditional containment failure probability

This step calculates the change in conditional containment failure probability (CCFP). Similar to Step 9 (Section 2.4.9) of this report, 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:

$$\Delta CCFP_{LINER10\text{-}15} = CCFP_{LINER1\text{-}15} - CCFP_{LINER1\text{-}10}$$

Where:

$\Delta CCFP_{LINER10\text{-}15}$ = the change in conditional containment failure probability from 1-in-10 years to 1-in-15 years given non-detected containment leakage

$CCFP_{LINER10}$ = conditional containment failure probability given 1-in-10 years ILRT interval and potential non-detected containment leakage [Step 7B]

$CCFP_{LINER15}$ = conditional containment failure probability given 1-in-15 years ILRT interval and potential non-detected containment leakage [Step 7B]

Therefore,

$\Delta CCFP_{LINER10\text{-}15}$ = 73.2% - 72.6%

$\Delta CCFP_{LINER10\text{-}15}$ = 0.6%

This change in $\Delta CCFP_{LINER10\text{-}15}$ 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 B.

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 1.22×10^{-8}/ry. This is below the Regulatory Guide 1.174 [5] acceptance criteria threshold of 10^{-7}/yr.

3. The age-adjusted corrosion impact in dose increase is estimated to be 4.1×10^{-3} person-rem/ry or 0.57% 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 0.6%.

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

FitzPatrick Containment Failure Probability Given Containment Liner Flaw

Pressure (psia)	Containment Liner Failure Probability	Drywell Floor Failure Probability
0	0.0000	0.0000
10	0.0006	0.0001
15	0.0008	0.0001
20	0.0010	0.0001
30	0.0017	0.0002
40	0.0028	0.0003
50	0.0046	0.0005
60	0.0077	0.0008
70	0.0128	0.0013
80	0.0214	0.0021
90	0.0357	0.0036
100	0.0596	0.0060
110	0.0994	0.0099
120	0.1659	0.0166
130	0.2768	0.0277
140	0.4618	0.0462
150	0.7706	0.0771
155	0.9954	0.0995

Figure B-1 - FitzPatrick Containment Failure Probability Given Containment Liner Flaw

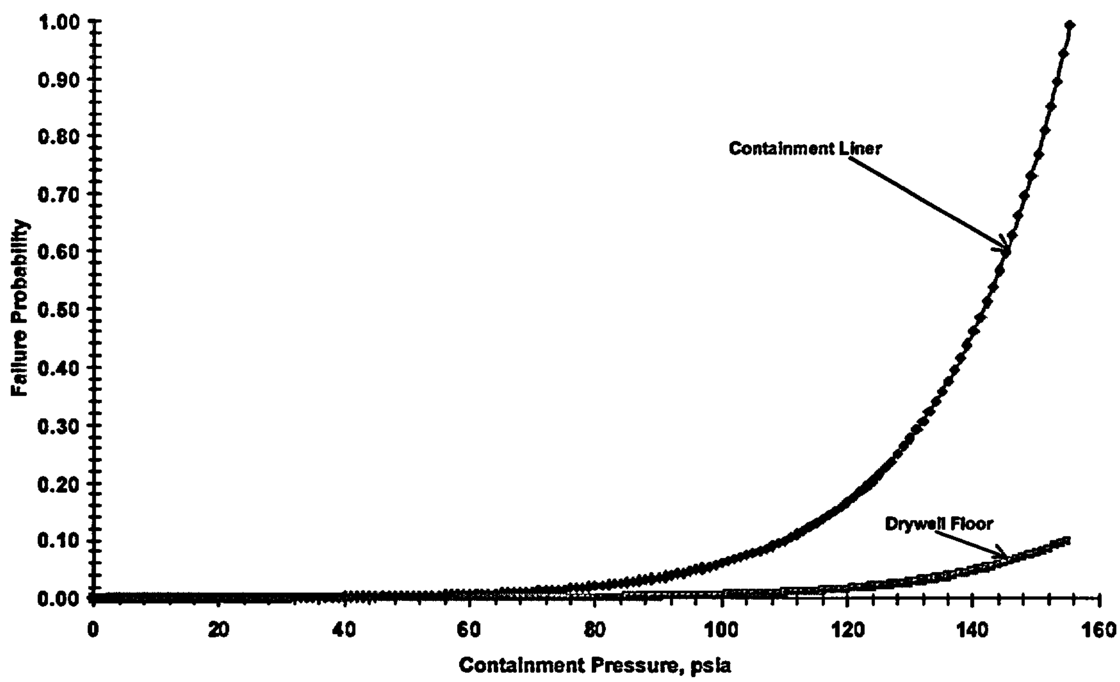


Table B-4

FitzPatrick 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 5-15 15	2.05×10^{-3} 5.19×10^{-3} 1.43×10^{-2}	1 5-15 15	2.05×10^{-3} 5.19×10^{-3} 1.43×10^{-2}	1 5-15 15	4.46×10^{-4} 1.30×10^{-3} 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-3 years) 9.68% (1-to-3 years)		0.18% (3-to-10 years) 1.04% (1-to-3 years) 2.42% (1-to-3 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 61.2 (ILRT) 100 120 155	0.001 0.008 0.060 0.166 0.995	20 61.2 (ILRT) 100 120 155	0.001 0.008 0.060 0.166 0.995	20 61.2 (ILRT) 100 120 155	0.000 0.0008 0.006 0.017 0.100
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)	0.00058% (3-to-10 years) 0.00340% (1-to-10 years) 0.00794% (1-to-15 years)		0.00582% (3-to-10 years) 0.03390% (1-to-10 years) 0.07938% (1-to-15 years)		0.00015% (3-to-10 years) 0.00085% (1-to-10 years) 0.00198% (1-to-15 years)	
Total Likelihood of Non-Detected Containment Leakage		0.00655% (3-to-10 years) 0.03815% (1-to-10 years) 0.08975% (1-to-15 years)					

Table B-5

Impact of Containment Steel Liner Corrosion on FitzPatrick 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	6.18×10^{-7}	2.91×10^{3}	1.80×10^{-3}	4.49×10^{-7}	2.91×10^{3}	1.31×10^{-3}	3.26×10^{-7}	2.91×10^{3}	9.52×10^{-4}
2	2.44×10^{-9}	6.79×10^{5}	1.66×10^{-3}	2.44×10^{-9}	6.79×10^{5}	1.66×10^{-3}	2.44×10^{-9}	6.79×10^{5}	1.66×10^{-3}
3a	6.59×10^{-8}	2.91×10^{4}	1.92×10^{-3}	2.20×10^{-7}	2.91×10^{4}	6.40×10^{-3}	3.29×10^{-7}	2.91×10^{4}	9.60×10^{-3}
3b	6.75×10^{-9}	1.02×10^{5}	6.88×10^{-4}	2.29×10^{-8}	1.02×10^{5}	2.34×10^{-3}	3.51×10^{-8}	1.02×10^{5}	3.58×10^{-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	1.74×10^{-7}	6.11×10^{5}	1.06×10^{-1}	1.74×10^{-7}	6.11×10^{5}	1.06×10^{-1}	1.74×10^{-7}	6.11×10^{5}	1.06×10^{-1}
7b	7.37×10^{-7}	3.96×10^{5}	2.91×10^{-1}	7.37×10^{-7}	3.96×10^{5}	2.91×10^{-1}	7.37×10^{-7}	3.96×10^{5}	2.91×10^{-1}
7c	5.82×10^{-8}	5.21×10^{5}	3.04×10^{-2}	5.82×10^{-8}	5.21×10^{5}	3.04×10^{-2}	5.82×10^{-8}	5.21×10^{5}	3.04×10^{-2}
7d	4.83×10^{-7}	3.06×10^{5}	1.48×10^{-1}	4.83×10^{-7}	3.06×10^{5}	1.48×10^{-1}	4.83×10^{-7}	3.06×10^{5}	1.48×10^{-1}
8	2.97×10^{-7}	4.46×10^{5}	1.32×10^{-1}	2.97×10^{-7}	4.46×10^{5}	1.32×10^{-1}	2.97×10^{-7}	4.46×10^{5}	1.32×10^{-1}
Total	2.44×10^{-6}		7.1×10^{-1}	2.44×10^{-6}		7.20×10^{-1}	2.44×10^{-6}		7.24×10^{-1}
ILRT Dose Rate from 3a and 3b			2.61×10^{-3} ($+1.63 \times 10^{-5}$)*			8.74×10^{-3} ($+9.48 \times 10^{-5}$)*			1.32×10^{-2} (+2.21E-04)*
% Of Total			0.365% (+0.0023%)*			1.2128% (+0.0130%)*			1.8199% (+0.0300%)*
Delta Dose Rate from 3a and 3b (10 to 15 yr)									4.09×10^{-3} (+0.0123%)*
LERF from 3b			6.75×10^{-9} ($+1.60 \times 10^{-10}$)*			2.29×10^{-8} ($+9.29 \times 10^{-10}$)*			3.51×10^{-8} ($+2.17 \times 10^{-9}$)*
Delta LERF (10 to 15 yr)									1.22×10^{-8} ($+1.24 \times 10^{-9}$)*
CCFP %			71.99% (+0.0065%)*			72.65% (+0.0381%)*			73.15% (+0.0889%)*
Delta CCFP % (10 to 15 yr)									0.60% (+0.0508%)*

* Denotes increase from original values presented in Steps 7, 8, and 9 (Section 2.4.7, 2.4.8 and 2.4.9) of this report.

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)
<u>Base Case</u> Doubles every 5 yrs	<u>Base Case</u> 0.8171%liner 0.0817%floor	<u>Base Case</u> 10%	<u>Base Case</u> 100%	<u>Base Case</u> 1.60×10^{-10}	<u>Base Case</u> 9.29×10^{-10}	<u>Base Case</u> 2.17×10^{-9}	<u>Base Case</u> **1.22×10^{-8}**
Doubles every 2 yrs	Base	Base	Base	4.57×10^{-11}	7.75×10^{-10}	4.50×10^{-9}	**1.47×10^{-8}**
Doubles every 10 yrs	Base	Base	Base	2.37×10^{-10}	3.25×10^{-10}	4.20×10^{-10}	**1.11×10^{-8}**
Base	Base	5%	Base	1.53×10^{-10}	6.20×10^{-11}	1.45×10^{-10}	**1.11×10^{-8}**
Base	Base	15%	Base	1.67×10^{-10}	1.45×10^{-10}	3.38×10^{-10}	**1.12×10^{-8}**
Base	0.166%liner[17] 0.017%floor[13]	Base	Base	3.25×10^{-11}	1.89×10^{-10}	4.42×10^{-10}	**1.12×10^{-8}**
Base	4.07% liner[18] 0.411%floor[14]	Base	Base	7.95×10^{-10}	4.63×10^{-9}	1.08×10^{-8}	**1.72×10^{-8}**
Lower Bound							
Doubles every 10 yrs	Base	5%	10%	4.62×10^{-12}	2.00×10^{-11}	3.66×10^{-11}	**1.10E-08**
Upper Bound							
Doubles every 2 yrs	Base	15%	100%	2.38×10^{-10}	4.03×10^{-9}	2.34×10^{-8}	**3.04E-08**

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

Attachment A

FitzPatrick IPE Level 2 Model Results to Support One-Time Extension

Table of Contents

A1.0 Introduction

This attachment presents the results of the FitzPatrick Level 2 IPE model to support the one-time exemption to the ten-year frequency of the performance-based Type A containment leakage-testing program for the Fitzpatrick plant. Specifically, the values of Tables 2-1, 2-2 and 2-3 are presented.

A2.0 Mean Internal Core Damage Frequency Contributions by Plant Damage States Results

Table 2-1 of this report presents the mean internal core damage frequency contributions by plant damage states. The table uses the plant damage states frequencies as reported in Table 4.4.2.2 (Section 4.4) of the FitzPatrick IPE [6]. The mean internal plant damage states frequencies are computed by taking the individual plant damage states frequencies and dividing by the point estimate core damage frequency value of 2.17 x 10^{6}/ry to obtain the percent contribution to core damage frequency for each plant damage state (column label % of CDF). The mean internal core damage frequency is then obtained by multiplying this result by the mean core damage frequency of 2.44 x 10^{-6}/ry (FitzPatrick IPE, Revision 1) [6].

These results are use in Table 2-2 of this report (the mean internal core damage frequency contributions by plant damage states)

The EXCEL spreadsheet data is as follows:

Mean Internal Core Damage Frequency Contributions by Plant Damage States Results

From Table 4.4.2.2 Rev.1	Frequency (/ry)	%CDF-PDS	PDS Frequency
TOTAL PDS1 FREQUENCY =	1.28E-08	0.0059	1.44E-08
TOTAL PDS2 FREQUENCY =	1.69E-07	0.0778	1.90E-07
TOTAL PDS3 FREQUENCY =	1.11E-06	0.5110	1.25E-06
TOTAL PDS4 FREQUENCY =	1.90E-07	0.0875	2.13E-07
TOTAL PDS5 FREQUENCY =	4.08E-08	0.0188	4.58E-08
TOTAL PDS6 FREQUENCY =	7.71E-08	0.0355	8.66E-08
TOTAL PDS7 FREQUENCY =	5.68E-09	0.0026	6.38E-09
TOTAL PDS8 FREQUENCY =	3.56E-08	0.0164	4.00E-08
TOTAL PDS9 FREQUENCY =	1.44E-08	0.0066	1.62E-08
TOTAL PDS10 FREQUENCY =	2.50E-07	0.1151	2.81E-07
TOTAL PDS11 FREQUENCY =	2.67E-07	0.1229	3.00E-07
Total=	**2.17E-06**	**1.00E+00**	**2.44E-06**

A3.0 Summary of FitzPatrick IPE LEVEL 2 Release Categories

Table 2-2 of this report presents a summary of FitzPatrick's IPE Level 2 release categories based on the mean internal core damage frequency of 2.44 x 10^{-6}/ry [6]. The results presented in Table 2-2 are computed in a two-step process. Step one multiplies the individual plant damage states frequencies by the release magnitudes split fractions generated by the Event Progress Analysis Code (EVNTRE) [30]. Step two, takes these results and sums the value for each of the nine-release category found in the Fitzpatrick IPE analysis.

Shown below are the EXCEL spreadsheets for each plant damage state release bins, followed by the summary of FitzPatrick's IPE Level 2 release categories based on the mean internal core damage frequency.

Individual Plant Damage States Release Categories

Release Category – Split Fractions	PDS-1	PSD-2	PDS-3	PDS-4	PDS-5	PDS-6
Late Low	0.000	0.000	0.000	0.040	0.013	0.006
Late Medium Low	0.313	0.313	0.279	0.085	0.293	0.180
L Medium High	0.000	0.000	0.000	0.000	0.000	0.000
Late High	0.013	0.013	0.010	0.095	0.024	0.242
Early Low	0.000	0.000	0.000	0.000	0.000	0.000
Early Medium Low	0.000	0.000	0.000	0.021	0.000	0.000
Early Medium High	0.245	0.245	0.279	0.145	0.239	0.107
Early High	0.000	0.000	0.000	0.522	0.047	0.374
No Containment Failure	0.429	0.429	0.432	0.094	0.384	0.091
Total=	1.000	1.000	1.000	1.000	1.000	1.000

Individual Plant Damage States Release Categories

Release Category – Split Fractions	PDS-7	PSD-8	PDS-9	PDS-10	PDS-11
Late Low	0.007	0.000	0.000	0.000	0.000
Late Medium Low	0.316	0.313	0.000	0.000	0.000
L Medium High	0.000	0.000	0.000	0.000	0.000
Late High	0.016	0.013	0.000	0.000	0.000
Early Low	0.000	0.000	0.000	0.000	0.000
Early Medium Low	0.000	0.000	0.885	0.887	0.788
Early Medium High	0.245	0.245	0.115	0.113	0.118
Early High	0.000	0.000	0.000	0.000	0.094
No Containment Failure	0.416	0.429	0.000	0.000	0.000
Total=	1.000	1.000	1.000	1.000	1.000

Individual Plant Damage States Release Categories

Release Category – Frequency/ry	PDS-1	PSD-2	PDS-3	PDS-4
Late Low	0.00E+00	0.00E+00	0.00E+00	8.62E-09
Late Medium Low	4.50E-09	5.94E-08	3.48E-07	1.81E-08
L Medium High	0.00E+00	0.00E+00	0.00E+00	8.37E-12
Late High	1.88E-10	2.49E-09	1.26E-08	2.03E-08
Early Low	0.00E+00	0.00E+00	0.00E+00	0.00E+00
Early Medium Low	0.00E+00	0.00E+00	0.00E+00	4.46E-09
Early Medium High	3.52E-09	4.65E-08	3.48E-07	3.09E-08
Early High	0.00E+00	0.00E+00	0.00E+00	1.11E-07
No Containment Failure	6.17E-09	8.14E-08	5.39E-07	2.00E-08
Total=	1.44E-08	1.90E-07	1.25E-06	2.14E-07

Individual Plant Damage States Release Categories

Release Category – Frequency/ry	PDS-5	PSD-6	PDS-7	PDS-8
Late Low	5.91E-10	5.01E-10	4.68E-11	0.00E+00
Late Medium Low	1.34E-08	1.56E-08	2.02E-09	1.25E-08
L Medium High	9.44E-12	7.59E-12	1.67E-12	0.00E+00
Late High	1.08E-09	2.10E-08	1.03E-10	5.20E-10
Early Low	0.00E+00	0.00E+00	0.00E+00	0.00E+00
Early Medium Low	2.07E-11	0.00E+00	0.00E+00	0.00E+00
Early Medium High	1.10E-08	9.27E-09	1.56E-09	9.80E-09
Early High	2.16E-09	3.24E-08	0.00E+00	0.00E+00
No Containment Failure	1.76E-08	7.88E-09	2.65E-09	1.72E-08
Total=	4.58E-08	8.66E-08	6.38E-09	4.00E-08

Individual Plant Damage States Release Categories

Release Category – Frequency/ry	PDS-9	PSD-10	PDS-11	TOTAL[19]
Late Low	0.00E+00	0.00E+00	0.00E+00	9.76E-09
Late Medium Low	0.00E+00	0.00E+00	0.00E+00	4.73E-07
L Medium High	0.00E+00	0.00E+00	0.00E+00	2.71E-11
Late High	0.00E+00	0.00E+00	0.00E+00	5.82E-08
Early Low	0.00E+00	0.00E+00	0.00E+00	0.00E+00
Early Medium Low	1.43E-08	2.49E-07	2.36E-07	5.04E-07
Early Medium High	1.86E-09	3.17E-08	3.54E-08	5.29E-07
Early High	0.00E+00	0.00E+00	2.82E-08	1.74E-07
No Containment Failure	0.00E+00	0.00E+00	0.00E+00	6.91E-07
Total=	1.62E-08	2.81E-07	3.00E-07	2.44E-06

[19] These are the values used in Table 2.

A4.0 Summary of FitzPatrick IPE LEVEL 2 Containment Failures

Table 2-3 of this report presents a summary of FitzPatrick's IPE Level 2 containment failures bins frequencies based on the mean internal core damage frequency of 2.44 x 10^{-6}/ry [6]. The results presented in Table 2-3 are computed as follows:

No Containment Failure Frequency = Frequency of No Containment Failure Release Category

No Containment Failure = 6.91E-07/ry

Early Drywell Failure Frequency = Frequency of Early High Release Category

Early Drywell Failure = 1.74E-07/ry

Early Wetwell Failure Frequency = (Frequency of Early Medium Low Release Category
+ Frequency of Early Medium High Release Category)
- (Frequency of ATWS Early Medium Low Release Category
+ Frequency of ATWS Early Medium High Release Category)

Early Wetwell Failure Frequency = (5.04E-07 + 5.29E-07)
- (1.43E-08 + 2.49E-07 + 1.86E-09 + 3.17E-08)

Early Wetwell Failure Frequency = 7.37E-07/ry

Late Drywell Failure Frequency = Frequency of Late High Release Category

Late Drywell Failure = 5.82E-08/ry

Late Wetwell Failure Frequency = Frequency of Late Low Release Category
+ Frequency of Late Medium Low Release Category
+ Frequency of Late Medium High Release Category

Late Wetwell Failure Frequency = 9.76E-09 + 4.73E-07 + 2.71E-11

Late Wetwell Failure Frequency = 4.83E-07

Bypass Failure Frequency = Frequency of ATWS Early Medium Low Release Category

+ Frequency of ATWS Early Medium High Release Category

Bypass Failure Frequency = 1.43E-08 + 2.49E-07 + 1.86E-09 + 3.17E-08

Bypass Failure Frequency = 2.97E-07/ry

A5.0 FitzPatrick Collapsed Accident Progression Bins Frequencies

Table 2-8 of this report presents the associated collapsed accident progression bins frequencies for FitzPatrick.

The results presented in Table 2-8 are determined by running the EVNTRE [30]. This code is used to compute the ten collapsed accident progression bins for use in mapping the FitzPatrick person-rem bins to those found in the NUREG/CR-4551 Peach Bottom, Unit 2 analysis [9 and 12]. The EXCEL spreadsheet results are summarized below and depicted in Figures A1 to A11.

Legend:

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

Collapsed Accident Progression Bins –Split Factions	PDS-1	PSD-2	PDS-3	PDS-4	PDS-5	PDS-6
VB, Early CF, WW, RPV pressure >200 psig at VB	0.000	0.000	0.0000	0.016	0.000	0.000
VB, Early CF, WW, RPV pressure <200 psig at VB	0.000	0.000	0.0000	0.000	0.000	0.000
VB, Early CF, DW, RPV pressure >200 psig at VB	0.000	0.000	0.0830	0.616	0.066	0.000
VB, Early CF, DW, RPV pressure <200 psig at VB	0.245	0.245	0.1961	0.053	0.221	0.481
VB, Late CF, WW	0.000	0.000	0.0000	0.001	0.003	0.000
VB, Late CF, DW	0.067	0.067	0.0534	0.103	0.071	0.252
VB, VENT	0.259	0.259	0.2352	0.117	0.255	0.175
VB, No CF	0.139	0.139	0.1950	0.028	0.115	0.027
No VB, No CF	0.290	0.290	0.2373	0.066	0.269	0.064
No CD, No CF	0.000	0.000	0.0000	0.000	0.000	0.000
Total=	1.000	1.000	1.000	1.000	1.000	1.000

Collapsed Accident Progression Bins –Split Factions	PDS-7	PSD-8	PDS-9	PDS-10	PDS-11
VB, Early CF, WW, RPV pressure >200 psig at VB	0.000	0.000	0.000	0.081	0.389
VB, Early CF, WW, RPV pressure <200 psig at VB	0.000	0.000	0.082	0.000	0.003
VB, Early CF, DW, RPV pressure >200 psig at VB	0.000	0.000	0.000	0.494	0.548
VB, Early CF, DW, RPV pressure <200 psig at VB	0.245	0.245	0.461	0.000	0.032
VB, Late CF, WW	0.001	0.000	0.000	0.000	0.000
VB, Late CF, DW	0.072	0.067	0.000	0.000	0.000
VB, VENT	0.267	0.259	0.457	0.425	0.027
VB, No CF	0.125	0.139	0.000	0.000	0.000
No VB, No CF	0.290	0.290	0.000	0.000	0.000
No CD, No CF	0.000	0.000	0.000	0.000	0.000
Total=	1.000	1.000	1.000	1.000	1.000

Collapsed Accident Progression Bins – Frequency/ry	PDS-1	PSD-2	PDS-3	PDS-4
VB, Early CF, WW, RPV pressure >200 psig at VB	0.00E+00	0.00E+00	0.00E+00	3.38E-09
VB, Early CF, WW, RPV pressure <200 psig at VB	0.00E+00	0.00E+00	0.00E+00	0.00E+00
VB, Early CF, DW, RPV pressure >200 psig at VB	0.00E+00	0.00E+00	1.03E-07	1.31E-07
VB, Early CF, DW, RPV pressure <200 psig at VB	3.53E-09	4.65E-08	2.44E-07	1.14E-08
VB, Late CF, WW	0.00E+00	0.00E+00	0.00E+00	1.47E-10
VB, Late CF, DW	9.62E-10	1.27E-08	6.66E-08	2.21E-08
VB, VENT	3.72E-09	4.92E-08	2.93E-07	2.50E-08
VB, No CF	2.00E-09	2.64E-08	2.43E-07	5.92E-09
No VB, No CF	4.17E-09	5.50E-08	2.96E-07	1.41E-08
No CD, No CF	0.00E+00	0.00E+00	0.00E+00	0.00E+00
Total=	1.44E-08	1.90E-07	1.25E-06	2.13E-07

Collapsed Accident Progression Bins – Frequency/ry	PDS-5	PSD-6	PDS-7	PDS-8
VB, Early CF, WW, RPV pressure >200 psig at VB	0.00E+00	0.00E+00	0.00E+00	0.00E+00
VB, Early CF, WW, RPV pressure <200 psig at VB	0.00E+00	0.00E+00	0.00E+00	0.00E+00
VB, Early CF, DW, RPV pressure >200 psig at VB	3.01E-09	0.00E+00	0.00E+00	0.00E+00
VB, Early CF, DW, RPV pressure <200 psig at VB	1.01E-08	4.17E-08	1.56E-09	9.80E-09
VB, Late CF, WW	1.35E-10	2.27E-11	4.78E-12	0.00E+00
VB, Late CF, DW	3.27E-09	2.18E-08	4.60E-10	2.67E-09
VB, VENT	1.17E-08	1.52E-08	1.70E-09	1.04E-08
VB, No CF	5.29E-09	2.36E-09	8.00E-10	5.56E-09
No VB, No CF	1.23E-08	5.52E-09	1.85E-09	1.16E-08
No CD, No CF	0.00E+00	0.00E+00	0.00E+00	0.00E+00
Total=	4.58E-08	8.66E-08	6.38E-09	4.00E-08

Collapsed Accident Progression Bins – Frequency/ry	PDS-9	PSD-10	PDS-11	Total
VB, Early CF, WW, RPV pressure >200 psig at VB	0.00E+00	2.27E-08	1.17E-07	1.43E-07
VB, Early CF, WW, RPV pressure <200 psig at VB	1.33E-09	0.00E+00	9.40E-10	2.27E-09

VB, Early CF, DW, RPV pressure >200 psig at VB	0.00E+00	1.39E-07	1.64E-07	5.41E-07
VB, Early CF, DW, RPV pressure <200 psig at VB	7.46E-09	0.00E+00	9.71E-09	3.86E-07
VB, Late CF, WW	0.00E+00	0.00E+00	0.00E+00	3.09E-10
VB, Late CF, DW	0.00E+00	0.00E+00	0.00E+00	1.31E-07
VB, VENT	7.39E-09	1.19E-07	8.20E-09	5.45E-07
VB, No CF	0.00E+00	0.00E+00	0.00E+00	2.92E-07
No VB, No CF	0.00E+00	0.00E+00	0.00E+00	4.00E-07
No CD, No CF	0.00E+00	0.00E+00	0.00E+00	0.00E+00
Total=	1.62E-08	2.81E-07	3.00E-07	2.44E-06

Figure A1 - JAF ILRT Evaluation PDS-1 Base Results

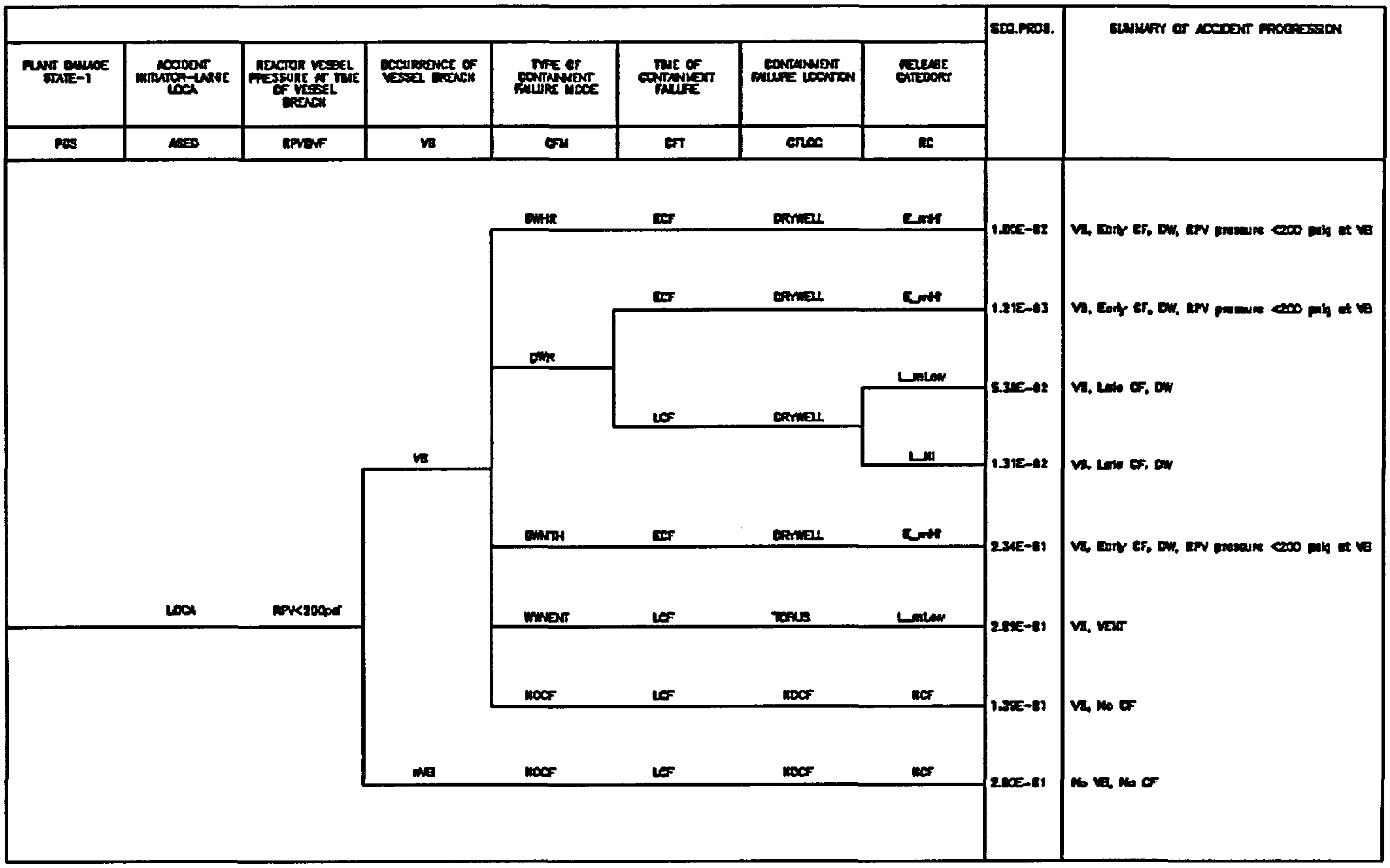


Figure A2 - JAF ILRT Evaluation PDS-2 Base Results

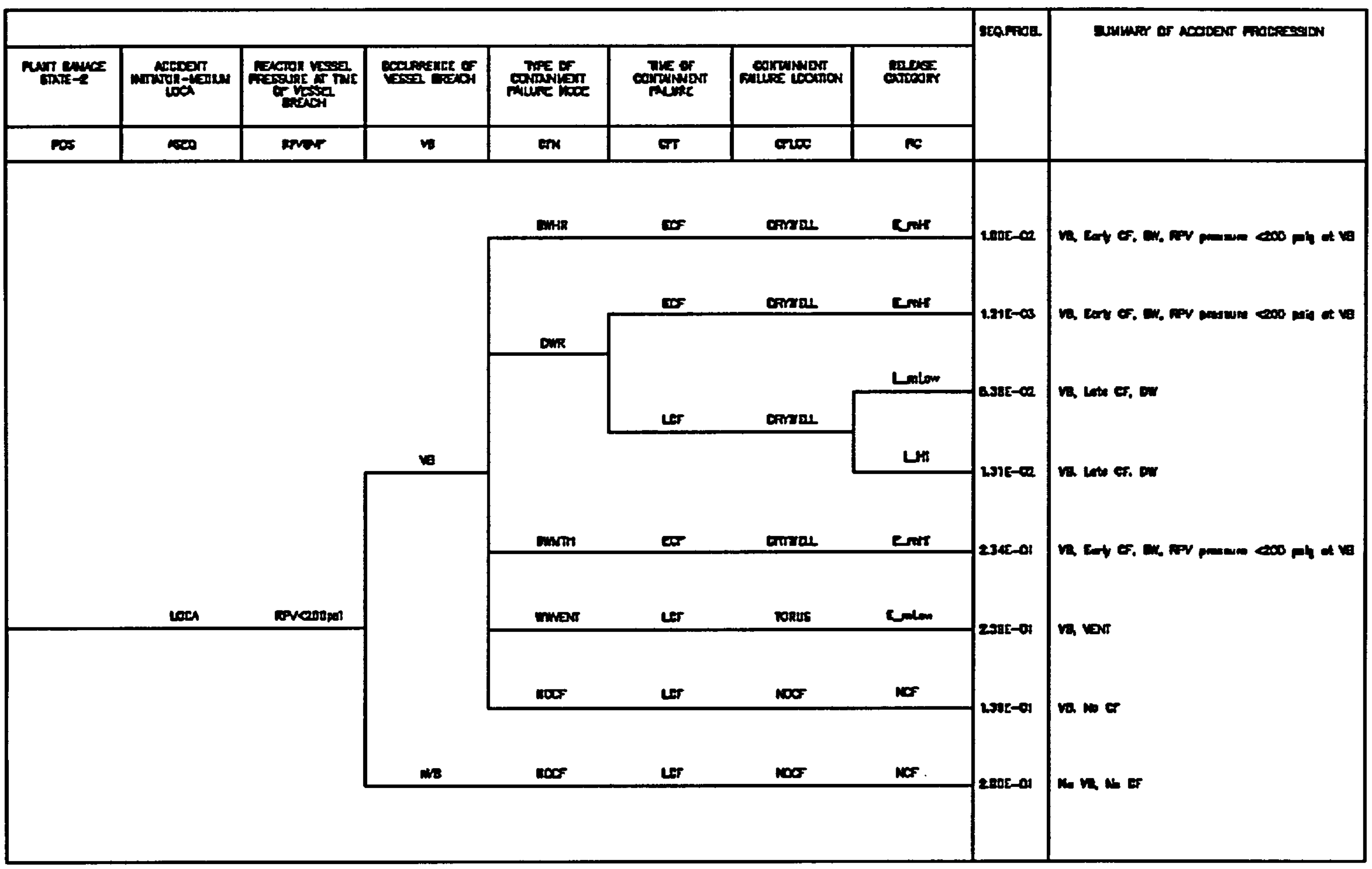


Figure A3 - JAF ILRT Evaluation PDS-3 Base Results

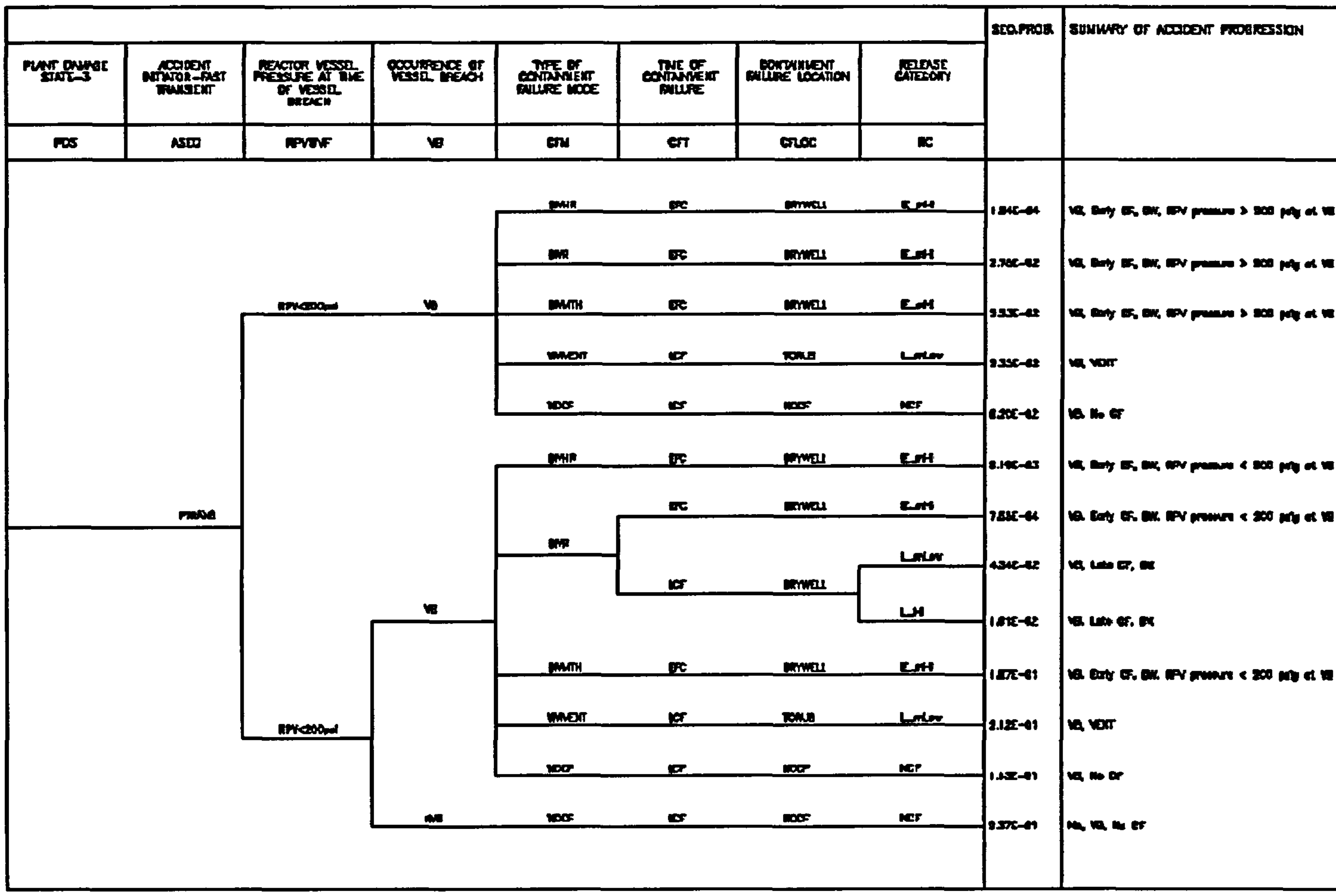


Figure A4 - JAF ILRT Evaluation PDS-4 Base Results

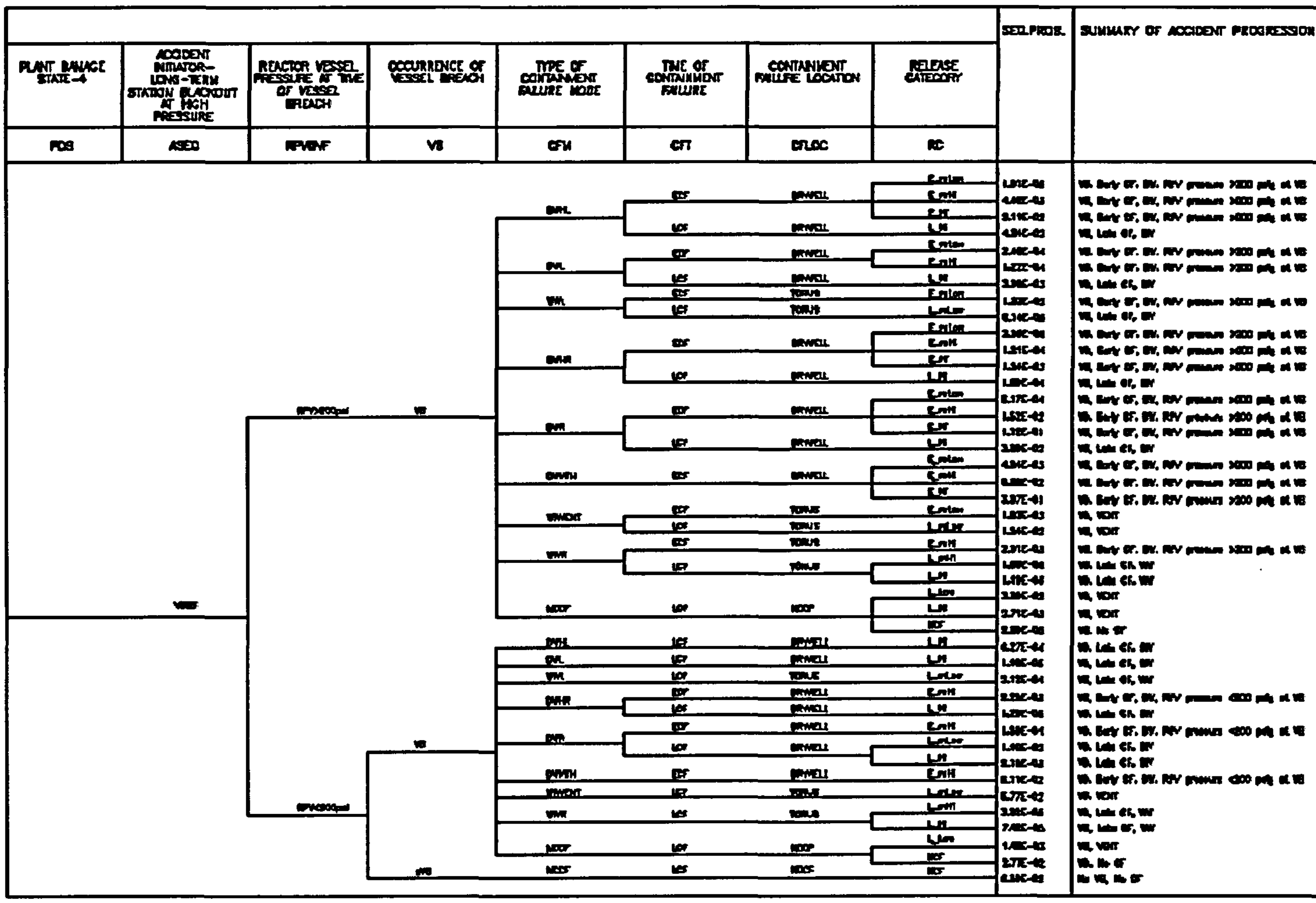


Figure A5 - JAF ILRT Evaluation PDS-5 Base Results

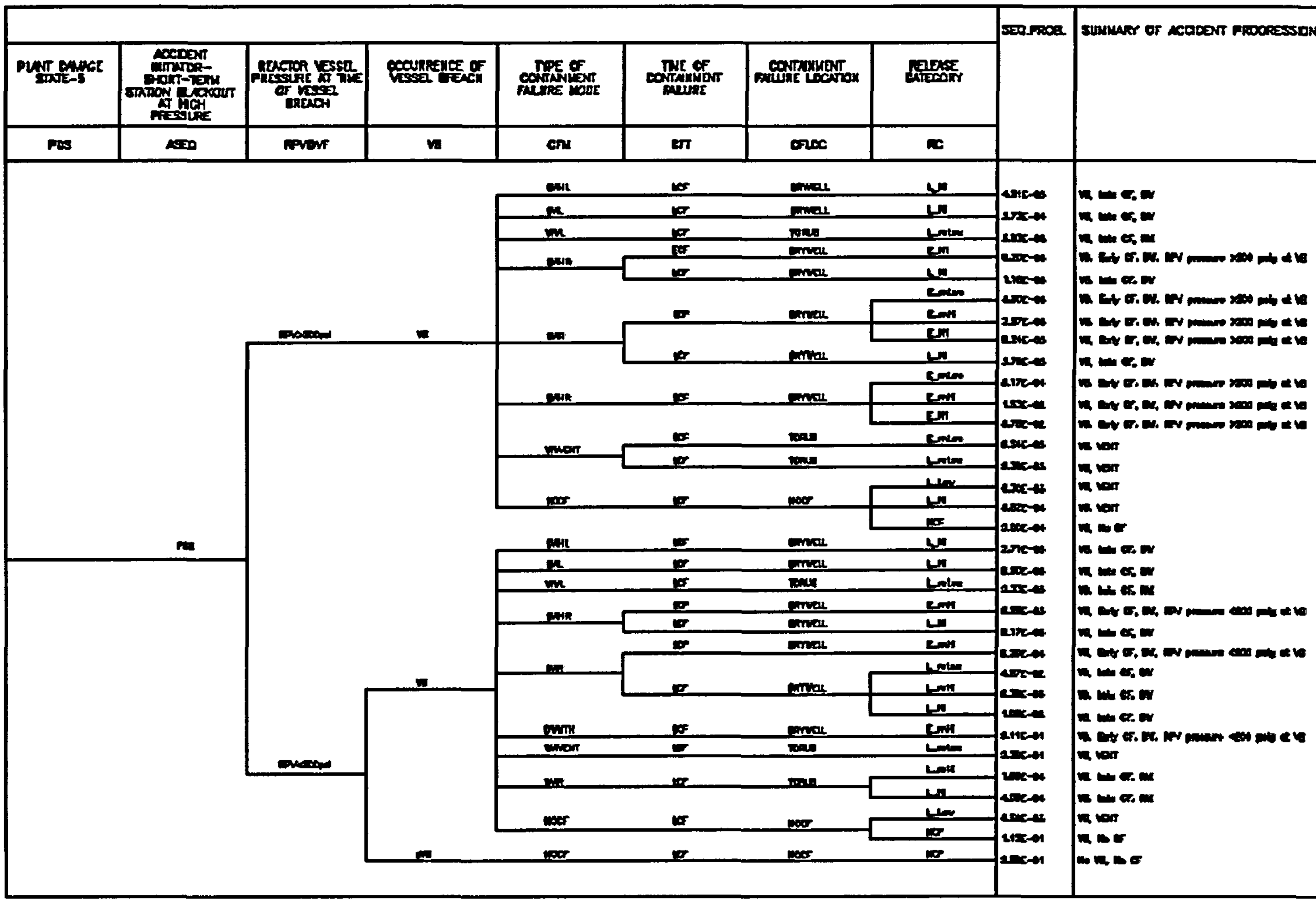


Figure A6 - JAF ILRT Evaluation PDS-6 Base Results

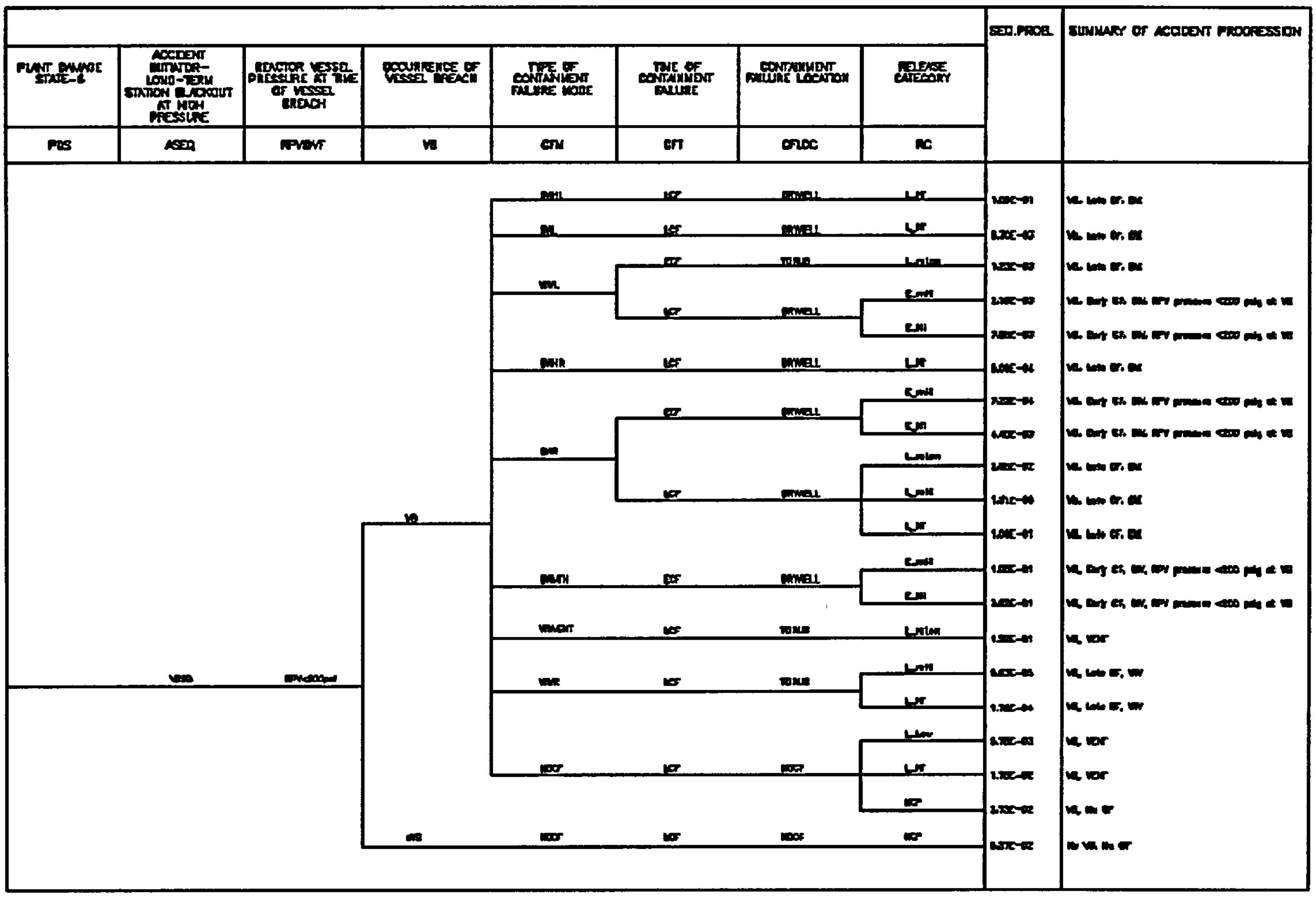


Figure A7 - JAF ILRT Evaluation PDS-7 Base Results

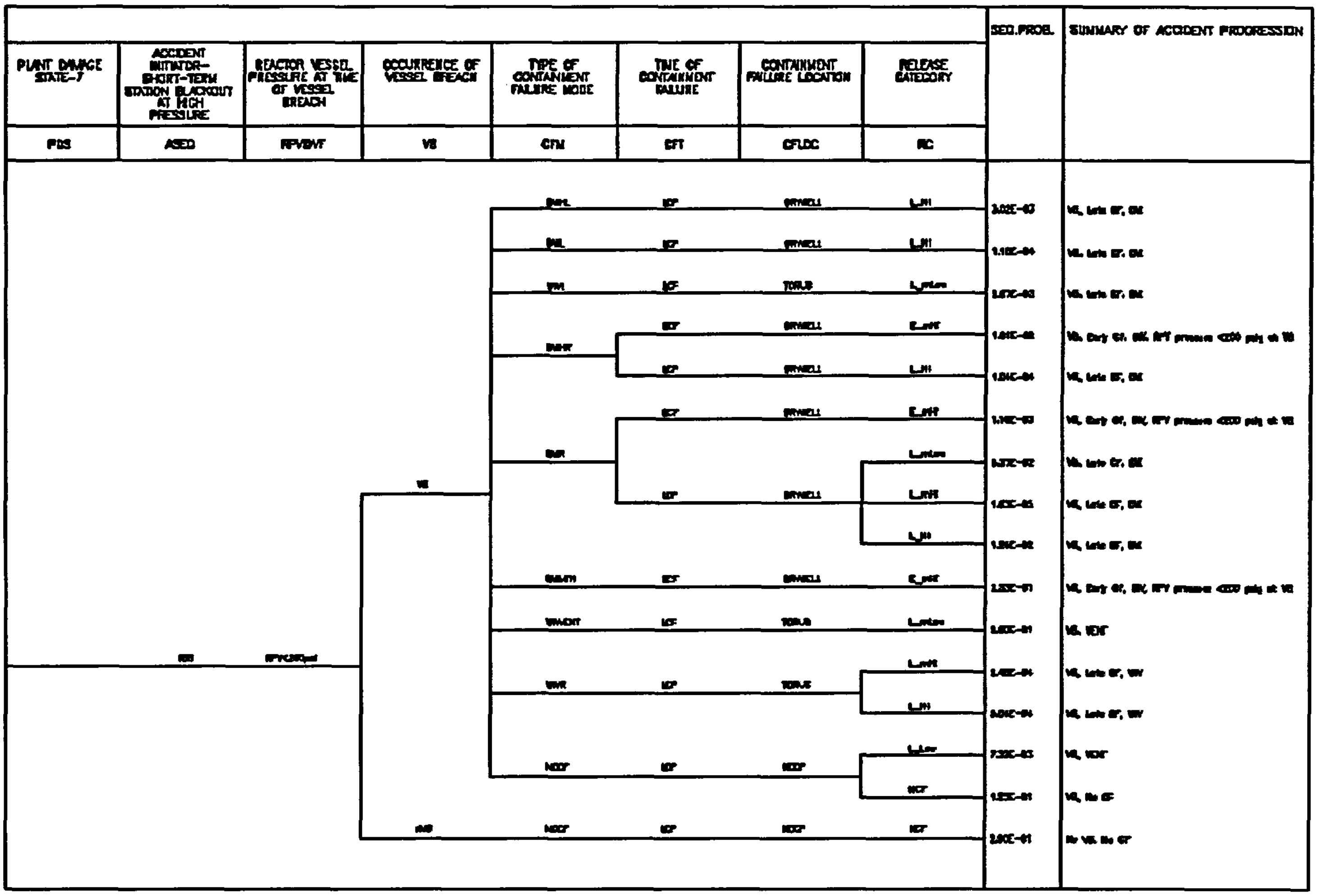


Figure A8 - JAF ILRT Evaluation PDS-8 Base Results

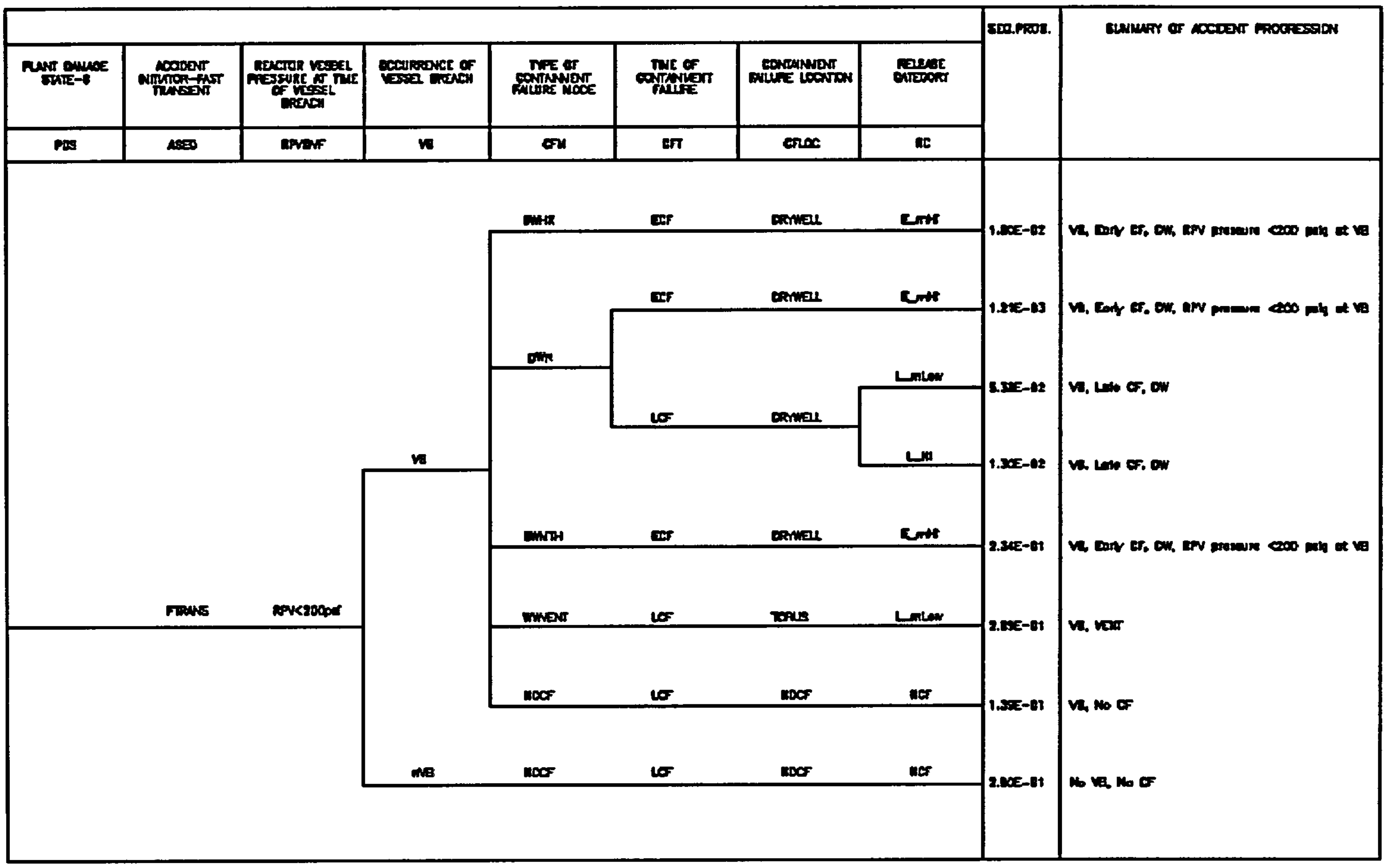


Figure A9 - JAF ILRT Evaluation PDS-9 Base Results

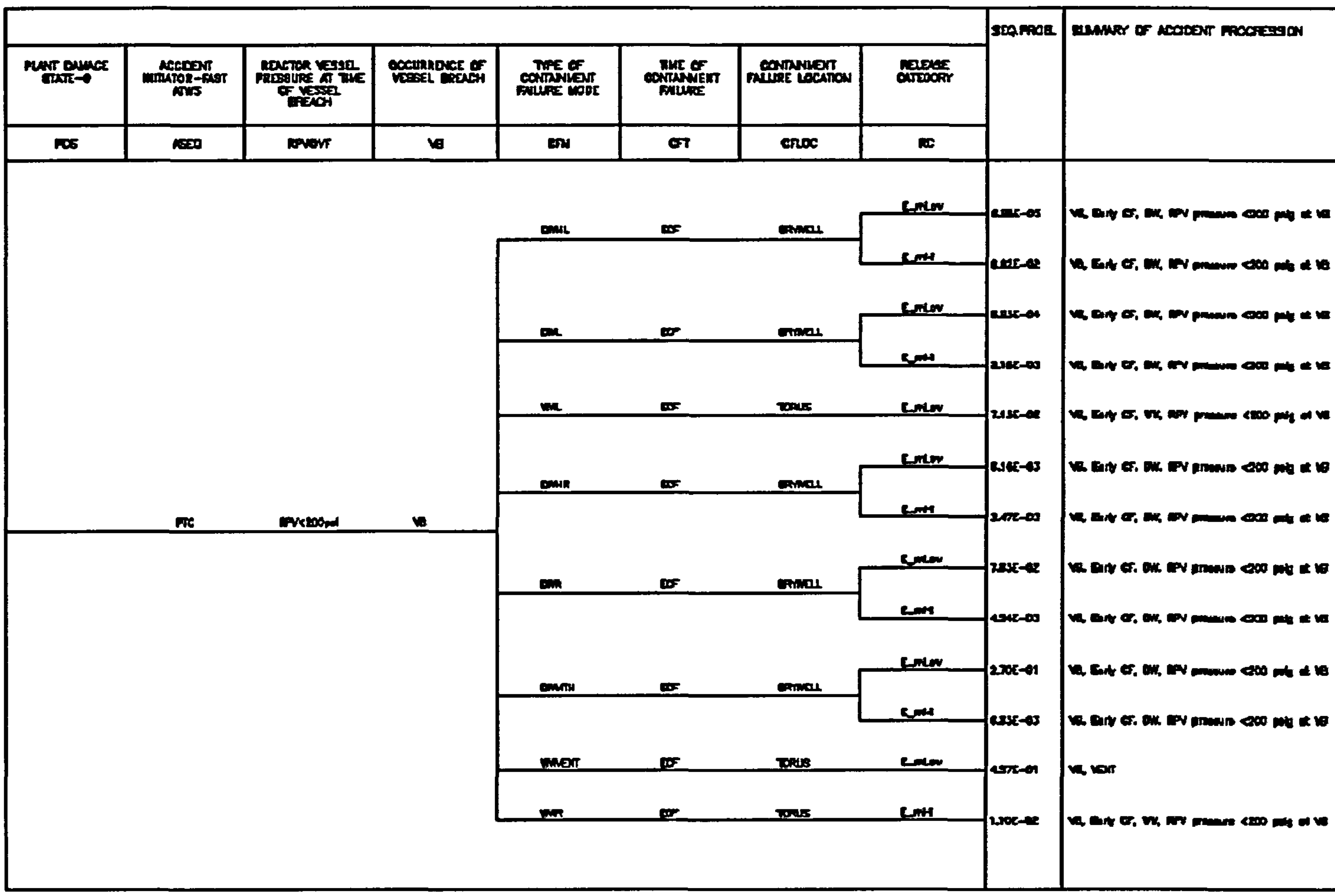


Figure A10 - JAF ILRT Evaluation PDS-10 Base Results

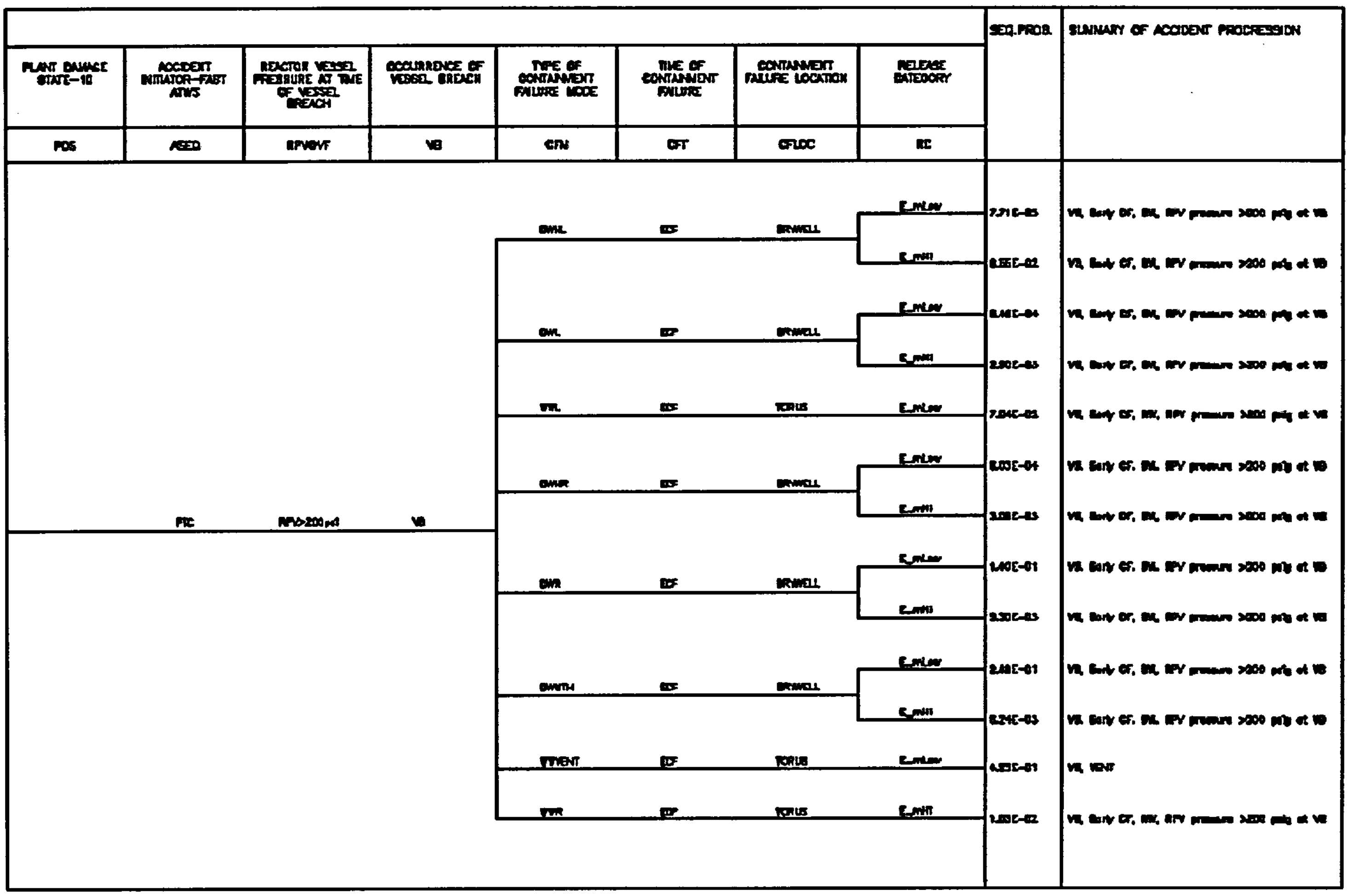
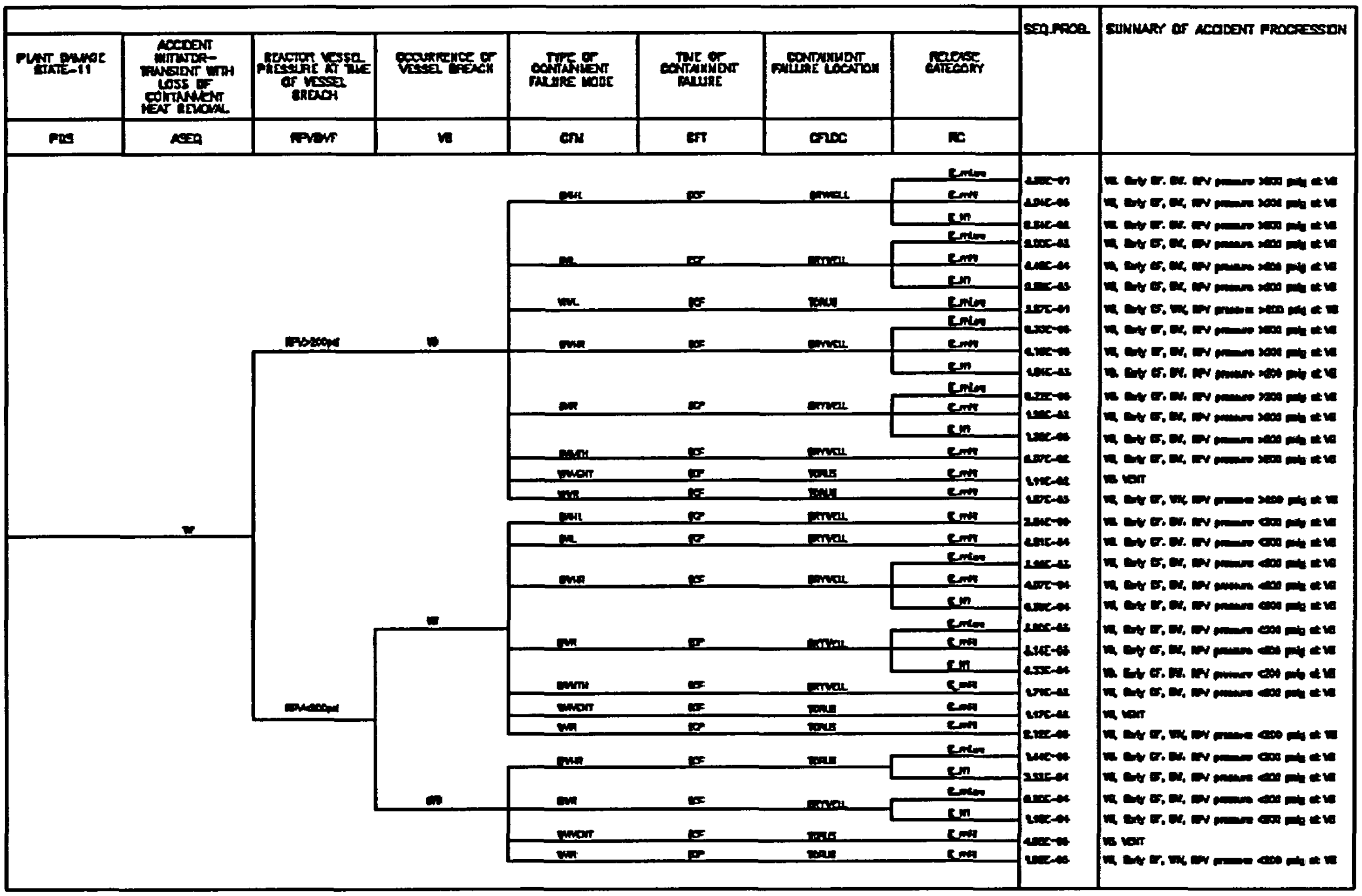


Figure A11 - JAF ILRT Evaluation PDS-11 Base Results



Attachment B

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

Table of Contents

B1.0 Introduction

This attachment presents the results of the FitzPatrick risk impact of containment liner corrosion during an extension of the ILRT interval. Seven 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.

B2.0 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	0.8171%	0.8171%	0.0817%
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%
	0.00017%	0.00167%	0.00004%	0.00187%
				0.00187%

Release type	FitzPatrick Dose Person-rem	CDF Frequency/ry	Case 3 in 10 yrs	Dose Person-rem/ry
1	2.91E+03	6.18E-07		1.80E-03
2	6.79E+05	2.44E-09	Corrosion Addition	1.66E-03
3a	2.91E+04	6.59E-08	0.00E+00	1.92E-03
3b	1.02E+05	6.63E-09	4.57E-11	6.77E-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	6.11E+05	1.74E-07		1.06E-01
7b	3.96E+05	7.37E-07		2.91E-01
7c	5.21E+05	5.82E-08		3.04E-02
7d	3.06E+05	4.83E-07		1.48E-01
8	4.46E+05	2.97E-07		1.32E-01
Total		**2.44E-06**		**0.7146**
			Risk Contribution:	0.36%
			From 3a and 3b:	2.60E-03
			3b LERF:	6.63E-09
			CCFP:	71.98%

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%
	0.0028%	0.0282%	0.0007%	0.03177%
				0.03177%

Release type	FitzPatrick Dose Person-rem	CDF Frequency/ry	Case 1 in 10 yrs	Dose Person-rem/ry
1	2.91E+03	4.49E-07		1.31E-03
2	6.79E+05	2.44E-09	Corrosion Addition	1.66E-03
3a	2.91E+04	2.20E-07	0.00E+00	6.40E-03
3b	1.02E+05	2.27E-08	7.75E-10	2.32E-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	6.11E+05	1.74E-07		1.06E-01
7b	3.96E+05	7.37E-07		2.91E-01
7c	5.21E+05	5.82E-08		3.04E-02
7d	3.06E+05	4.83E-07		1.48E-01
8	4.46E+05	2.97E-07		1.32E-01
Total		2.44E-06		0.7203
			Risk Contribution:	1.21%
			From 3a and 3b:	8.72E-03
			3b LERF:	2.27E-08

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%
	0.0164%	0.1640%	0.0041%	0.18447%
				0.18447%

Release type	FitzPatrick Dose Person-rem	CDF Frequency/ry	Case 1 in 15 yrs	Dose Person-rem/ry
1	2.91E+03	3.24E-07		9.45E-04
2	6.79E+05	2.44E-09	Corrosion Addition	1.66E-03
3a	2.91E+04	3.29E-07	0.00E+00	9.60E-03
3b	1.02E+05	3.74E-08	4.50E-09	3.82E-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	6.11E+05	1.74E-07		1.06E-01
7b	3.96E+05	7.37E-07		2.91E-01
7c	5.21E+05	5.82E-08		3.04E-02
7d	3.06E+05	4.83E-07		1.48E-01
8	4.46E+05	2.97E-07		1.32E-01
Total		2.44E-06		0.7246
			Risk Contribution:	1.85%
			From 3a and 3b:	1.34E-02
			3b LERF:	3.74E-08

Other Pertinent Risk Metrics

10 to 15 Increase (Person-rem/ry):	4.34E-03
3 to 15 Increase (Person-rem/ry):	9.97E-03
10 to 15 Delta-LERF:	1.47E-08
3 to 15 Delta-LERF:	3.08E-08
10 to 15 Delta-CCFP:	0.60%
3 to 15 Delta-CCFP:	1.26%
3 to 15 Delta-LERF from Corrosion:	3.08E-08
10 to 15 Delta-LERF from Corrosion:	1.47E-08
Increase in LERF (ILRT 3-to-15 years)	5.32E-09

B3.0 Sensitivity Case 2 - Flaw Rate Doubles Every 10 Years

<u>3-in-10 years</u>

From Estimated Change

	Drywell/Torus	Inaccessible DW/Torus	Drywell Floor	Total
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%	
<u>Other Assumptions:</u>				
Containment Breach	0.8171%	0.8171%	0.0817%	
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				
	0.00000%	0.00000%	0.00000%	0.00000%
	0.00086%	0.00865%	0.00022%	0.00973%
				0.00973%

Release type	FitzPatrick Dose Person-rem	CDF Frequency/ry	Case 3 in 10 yrs	Dose Person-rem/ry
1	2.91E+03	6.18E-07		1.80E-03
2	6.79E+05	2.44E-09	Corrosion Addition	1.66E-03
3a	2.91E+04	6.59E-08	0.00E+00	1.92E-03
3b	1.02E+05	6.83E-09	2.37E-10	6.96E-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	6.11E+05	1.74E-07		1.06E-01
7b	3.96E+05	7.37E-07		2.91E-01
7c	5.21E+05	5.82E-08		3.04E-02
7d	3.06E+05	4.83E-07		1.48E-01
8	4.46E+05	2.97E-07		1.32E-01
Total		**2.44E-06**		**0.7147**
			Risk Contribution:	0.37%
			From 3a and 3b:	2.62E-03
			3b LERF:	6.83E-09
			CCFP:	71.99%

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%
	0.0037%	0.0086%	0.0009%	0.01333%
				0.01333%

Release type	FitzPatrick Dose Person-rem	CDF Frequency/ry	Case 1 in 10 yrs	Dose Person-rem/ry
1	2.91E+03	4.49E-07		1.31E-03
2	6.79E+05	2.44E-09	Corrosion Addition	1.66E-03
3a	2.91E+04	2.20E-07	0.00E+00	6.40E-03
3b	1.02E+05	2.23E-08	3.25E-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	6.11E+05	1.74E-07		1.06E-01
7b	3.96E+05	7.37E-07		2.91E-01
7c	5.21E+05	5.82E-08		3.04E-02
7d	3.06E+05	4.83E-07		1.48E-01
8	4.46E+05	2.97E-07		1.32E-01
Total		2.44E-06		0.7202
			Risk Contribution:	1.20%
			From 3a and 3b:	8.67E-03
			3b LERF:	2.23E-08

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%
	0.0069%	0.0086%	0.0017%	0.01721%
				0.01721%

Release type	FitzPatrick Dose Person-rem	CDF Frequency/ry	Case 1 in 15 yrs	Dose Person-rem/ry
1	2.91E+03	3.28E-07		9.57E-04
2	6.79E+05	2.44E-09	Corrosion Addition	1.66E-03
3a	2.91E+04	3.29E-07	0.00E+00	9.60E-03
3b	1.02E+05	3.34E-08	4.20E-10	3.40E-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	6.11E+05	1.74E-07		1.06E-01
7b	3.96E+05	7.37E-07		2.91E-01
7c	5.21E+05	5.82E-08		3.04E-02
7d	3.06E+05	4.83E-07		1.48E-01
8	4.46E+05	2.97E-07		1.32E-01
Total		2.44E-06		0.7242
			Risk Contribution:	1.80%
			From 3a and 3b:	1.30E-02
			3b LERF:	3.34E-08

Other Pertinent Risk Metrics:

10 to 15 Increase (Person-rem/ry):	3.98E-03
3 to 15 Increase (Person-rem/ry):	9.54E-03
10 to 15 Delta-LERF:	1.11E-08
3 to 15 Delta-LERF:	2.65E-08
10 to 15 Delta-CCFP:	0.45%
3 to 15 Delta-CCFP:	1.09%
3 to 15 Delta-LERF from Corrosion:	2.65E-08
10 to 15 Delta-LERF from Corrosion:	1.11E-08
Increase in LERF (ILRT 3-to-15 years)	9.83E-10

B4.0 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	0.8171%	0.8171%	0.0817%
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%	0.0000%
	0.0003%	0.0058%	0.0001%	0.0063%
				0.0063%

Release type	FitzPatrick Dose Person-rem	CDF Frequency/ry	Case 3 in 10 yrs	Dose Person-rem/ry
1	2.91E+03	6.18E-07		1.80E-03
2	6.79E+05	2.44E-09	Corrosion Addition	1.66E-03
3a	2.91E+04	6.59E-08	0.00E+00	1.92E-03
3b	1.02E+05	6.74E-09	1.53E-10	6.88E-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	6.11E+05	1.74E-07		1.06E-01
7b	3.96E+05	7.37E-07		2.91E-01
7c	5.21E+05	5.82E-08		3.04E-02
7d	3.06E+05	4.83E-07		1.48E-01
8	4.46E+05	2.97E-07		1.32E-01
Total		2.44E-06		0.7147
			Risk Contribution:	0.36%
			From 3a and 3b:	2.61E-03
			3b LERF:	6.74E-09
			CCFP:	71.99%

1-in-10 years

Increases to 3a and 3b Frequencies	Drywell/Torus	Inaccessible DW/Torus	Drywell Floor	Total
	0.0000%	0.0000%	0.0000%	0.0000%
	0.0017%	0.0339%	0.0008%	0.0025%
				0.0025%

Release type	FitzPatrick Dose Person-rem	CDF Frequency/ry	Case 1 in 10 yrs	Dose Person-rem/ry
1	2.91E+03	4.49E-07		1.31E-03
2	6.79E+05	2.44E-09	Corrosion Addition	1.66E-03
3a	2.91E+04	2.20E-07	0.00E+00	6.40E-03
3b	1.02E+05	2.20E-08	6.20E-11	2.25E-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	6.11E+05	1.74E-07		1.06E-01
7b	3.96E+05	7.37E-07		2.91E-01
7c	5.21E+05	5.82E-08		3.04E-02
7d	3.06E+05	4.83E-07		1.48E-01
8	4.46E+05	2.97E-07		1.32E-01
Total		2.44E-06		0.7202
			Risk Contribution:	1.20%
			From 3a and 3b:	8.65E-03
			3b LERF:	2.20E-08

1-in-15 years

Increases to 3a and 3b Frequencies	Drywell/Torus	Inaccessible DW/Torus	Drywell Floor	Total
	0.0000%	0.0000%	0.0000%	0.0000%
	0.0040%	0.0791%	0.0020%	0.0059%
				0.0059%

Release type	FitzPatrick Dose Person-rem	CDF Frequency/ry	Case 1 in 15 yrs	Dose Person-rem/ry
1	2.91E+03	3.29E-07		9.58E-04
2	6.79E+05	2.44E-09	Corrosion Addition	1.66E-03
3a	2.91E+04	3.29E-07	0.00E+00	9.60E-03
3b	1.02E+05	3.31E-08	1.45E-10	3.38E-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	6.11E+05	1.74E-07		1.06E-01
7b	3.96E+05	7.37E-07		2.91E-01
7c	5.21E+05	5.82E-08		3.04E-02
7d	3.06E+05	4.83E-07		1.48E-01
8	4.46E+05	2.97E-07		1.32E-01
Total		2.44E-06		0.7242
			Risk Contribution:	1.79%
			From 3a and 3b:	1.30E-02
			3b LERF:	3.31E-08

Other Pertinent Risk Metrics

10 to 15 Increase (Person-rem/ry):	3.98E-03
3 to 15 Increase (Person-rem/ry):	9.52E-03
10 to 15 Delta-LERF:	1.11E-08
3 to 15 Delta-LERF:	2.63E-08
10 to 15 Delta-CCFP:	0.45%
3 to 15 Delta-CCFP:	1.08%
3 to 15 Delta-LERF from Corrosion:	2.63E-08
10 to 15 Delta-LERF from Corrosion:	1.11E-08
Increase in LERF (ILRT 3-to-15 years)	3.59E-10

B5.0 Sensitivity Case 4 - 15% Visual Inspection Failures

<u>3-in-10 years</u>

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	0.8171%	0.8171%	0.0817%
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%	0.0000%
	0.0009%	0.0058%	0.0001%	0.0068%
				0.0068%

Release type	FitzPatrick Dose Person-rem	CDF Frequency/ry	Case 3 in 10 yrs	Dose Person-rem/ry
1	2.91E+03	6.18E-07		1.80E-03
2	6.79E+05	2.44E-09	Corrosion Addition	1.66E-03
3a	2.91E+04	6.59E-08	0.00E+00	1.92E-03
3b	1.02E+05	6.75E-09	1.67E-10	6.89E-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	6.11E+05	1.74E-07		1.06E-01
7b	3.96E+05	7.37E-07		2.91E-01
7c	5.21E+05	5.82E-08		3.04E-02
7d	3.06E+05	4.83E-07		1.48E-01
8	4.46E+05	2.97E-07		1.32E-01
Total		**2.44E-06**		**0.7147**

Risk Contribution:	0.37%
From 3a and 3b:	2.61E-03
3b LERF:	6.75E-09
CCFP:	71.99%

1-in-10 years

Increases to 3a and 3b Frequencies	Drywell/Torus	Inaccessible DW/Torus	Drywell Floor	Total
	0.0000%	0.0000%	0.0000%	0.0000%
	0.0051%	0.0339%	0.0008%	0.0059%
				0.0059%

Release type	FitzPatrick Dose Person-rem	CDF Frequency/ry	Case 1 in 10 yrs	Dose Person-rem/ry
1	2.91E+03	4.49E-07		1.31E-03
2	6.79E+05	2.44E-09	Corrosion Addition	1.66E-03
3a	2.91E+04	2.20E-07	0.00E+00	6.40E-03
3b	1.02E+05	2.21E-08	1.45E-10	2.25E-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	6.11E+05	1.74E-07		1.06E-01
7b	3.96E+05	7.37E-07		2.91E-01
7c	5.21E+05	5.82E-08		3.04E-02
7d	3.06E+05	4.83E-07		1.48E-01
8	4.46E+05	2.97E-07		1.32E-01
Total		2.44E-06		0.7202
			Risk Contribution:	1.20%
			From 3a and 3b:	8.66E-03
			3b LERF:	2.21E-08

1-in-15 years

Increases to 3a and 3b Frequencies	Drywell/Torus	Inaccessible DW/Torus	Drywell Floor	Total
	0.0000%	0.0000%	0.0000%	0.0000%
	0.0119%	0.0791%	0.0020%	0.0138%
				0.0138%

Release type	FitzPatrick Dose Person-rem	CDF Frequency/ry	Case 1 in 15 yrs	Dose Person-rem/ry
1	2.91E+03	3.28E-07		9.57E-04
2	6.79E+05	2.44E-09	Corrosion Addition	1.66E-03
3a	2.91E+04	3.29E-07	0.00E+00	9.60E-03
3b	1.02E+05	3.33E-08	3.38E-10	3.39E-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	6.11E+05	1.74E-07		1.06E-01
7b	3.96E+05	7.37E-07		2.91E-01
7c	5.21E+05	5.82E-08		3.04E-02
7d	3.06E+05	4.83E-07		1.48E-01
8	4.46E+05	2.97E-07		1.32E-01
Total		2.44E-06		0.7242
			Risk Contribution:	1.79%
			From 3a and 3b:	1.30E-02
			3b LERF:	3.33E-08

Other Pertinent Risk Metrics:

10 to 15 Increase (Person-rem/ry):	3.99E-03
3 to 15 Increase (Person-rem/ry):	9.54E-03
10 to 15 Delta-LERF:	1.12E-08
3 to 15 Delta-LERF:	2.65E-08
10 to 15 Delta-CCFP:	0.46%
3 to 15 Delta-CCFP:	1.09%
3 to 15 Delta-LERF from Corrosion:	2.65E-08
10 to 15 Delta-LERF from Corrosion:	1.12E-08
Increase in LERF (ILRT 3-to-15 years)	6.49E-10

B6.0 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	0.1663%	0.1663%	0.0166%
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%
	0.00012%	0.00118%	0.00003%	0.00133%
				0.00133%

Release type	FitzPatrick Dose Person-rem	CDF Frequency/ry	Case 3 in 10 yrs	Dose Person-rem/ry
1	2.91E+03	6.18E-07		1.80E-03
2	6.79E+05	2.44E-09	Corrosion Addition	1.66E-03
3a	2.91E+04	6.59E-08	0.00E+00	1.92E-03
3b	1.02E+05	6.62E-09	3.25E-11	6.75E-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	6.11E+05	1.74E-07		1.06E-01
7b	3.96E+05	7.37E-07		2.91E-01
7c	5.21E+05	5.82E-08		3.04E-02
7d	3.06E+05	4.83E-07		1.48E-01
8	4.46E+05	2.97E-07		1.32E-01
Total		2.44E-06		0.7146
			Risk Contribution:	0.3632%
			From 3a and 3b:	2.60E-03
			3b LERF:	6.62E-09
			CCFP:	71.98%

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%
	0.00069%	0.00689%	0.00017%	0.00775%
				0.00775%

Release type	FitzPatrick Dose Person-rem	CDF Frequency/ry	Case 1 in 10 yrs	Dose Person-rem/ry
1	2.91E+03	4.49E-07		1.31E-03
2	6.79E+05	2.44E-09	Corrosion Addition	1.66E-03
3a	2.91E+04	2.20E-07	0.00E+00	6.40E-03
3b	1.02E+05	2.21E-08	1.89E-10	2.26E-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	6.11E+05	1.74E-07		1.06E-01
7b	3.96E+05	7.37E-07		2.91E-01
7c	5.21E+05	5.82E-08		3.04E-02
7d	3.06E+05	4.83E-07		1.48E-01
8	4.46E+05	2.97E-07		1.32E-01
Total		2.44E-06		0.7202
			Risk Contribution:	1.2024%
			From 3a and 3b:	8.66E-03
			3b LERF:	2.21E-08

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%
	0.00161%	0.01610%	0.00040%	0.01811%
				0.01811%

Release type	FitzPatrick Dose Person-rem	CDF Frequency/ry	Case 1 in 15 yrs	Dose Person-rem/ry
1	2.91E+03	3.28E-07		9.57E-04
2	6.79E+05	2.44E-09	Corrosion Addition	1.66E-03
3a	2.91E+04	3.29E-07	0.00E+00	9.60E-03
3b	1.02E+05	3.34E-08	4.42E-10	3.41E-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	6.11E+05	1.74E-07		1.06E-01
7b	3.96E+05	7.37E-07		2.91E-01
7c	5.21E+05	5.82E-08		3.04E-02
7d	3.06E+05	4.83E-07		1.48E-01
8	4.46E+05	2.97E-07		1.32E-01
Total		2.44E-06		0.7242
			Risk Contribution:	1.7959%
			From 3a and 3b:	1.30E-02
			3b LERF:	3.34E-08

Other Pertinent Risk Metrics:

10 to 15 Increase (Person-rem/ry):	3.99E-03
3 to 15 Increase (Person-rem/ry):	9.56E-03
10 to 15 Delta-LERF:	1.12E-08
3 to 15 Delta-LERF:	2.68E-08
10 to 15 Delta-CCFP:	0.460%
3 to 15 Delta-CCFP:	1.10%
3 to 15 Delta-LERF from Corrosion:	2.68E-08
10 to 15 Delta-LERF from Corrosion:	1.12E-08
Increase in LERF (ILRT 3-to-15 years)	6.64E-10

B7.0 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	4.0703%	4.0703%	0.4070%
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%
	0.00290%	0.02896%	0.00072%	0.03258%
				0.03258%

Release type	FitzPatrick Dose Person-rem	CDF Frequency/ry	Case 3 in 10 yrs	Dose Person-rem/ry
1	2.91E+03	6.18E-07		1.80E-03
2	6.79E+05	2.44E-09	Corrosion Addition	1.66E-03
3a	2.91E+04	6.59E-08	0.00E+00	1.92E-03
3b	1.02E+05	7.38E-09	7.95E-10	7.53E-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	6.11E+05	1.74E-07		1.06E-01
7b	3.96E+05	7.37E-07		2.91E-01
7c	5.21E+05	5.82E-08		3.04E-02
7d	3.06E+05	4.83E-07		1.48E-01
8	4.46E+05	2.97E-07		1.32E-01
Total		**2.44E-06**		**0.7147**

Risk Contribution:	0.3740%
From 3a and 3b:	2.67E-03
3b LERF:	7.38E-09
CCFP:	72.01%

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%
	0.01687%	0.16867%	0.00422%	0.18976%
				0.18976%

Release type	FitzPatrick Dose Person-rem	CDF Frequency/ry	Case 1 in 10 yrs	Dose Person-rem/ry
1	2.91E+03	4.45E-07		1.30E-03
2	6.79E+05	2.44E-09	Corrosion Addition	1.66E-03
3a	2.91E+04	2.20E-07	0.00E+00	6.40E-03
3b	1.02E+05	2.66E-08	4.63E-09	2.71E-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	6.11E+05	1.74E-07		1.06E-01
7b	3.96E+05	7.37E-07		2.91E-01
7c	5.21E+05	5.82E-08		3.04E-02
7d	3.06E+05	4.83E-07		1.48E-01
8	4.46E+05	2.97E-07		1.32E-01
Total		2.44E-06		0.7207
			Risk Contribution:	1.2646%
			From 3a and 3b:	9.11E-03
			3b LERF:	2.66E-08

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%
	0.03940%	0.39396%	0.00985%	0.44321%
				0.44321%

Release type	FitzPatrick Dose Person-rem	CDF Frequency/ry	Case 1 in 15 yrs	Dose Person-rem/ry
1	2.91E+03	3.18E-07		9.26E-04
2	6.79E+05	2.44E-09	Corrosion Addition	1.66E-03
3a	2.91E+04	3.29E-07	0.00E+00	9.60E-03
3b	1.02E+05	4.38E-08	1.08E-08	4.46E-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	6.11E+05	1.74E-07		1.06E-01
7b	3.96E+05	7.37E-07		2.91E-01
7c	5.21E+05	5.82E-08		3.04E-02
7d	3.06E+05	4.83E-07		1.48E-01
8	4.46E+05	2.97E-07		1.32E-01
Total		2.44E-06		0.7252
			Risk Contribution:	1.9393%
			From 3a and 3b:	1.41E-02
			3b LERF:	4.38E-08

Other Pertinent Risk Metrics

10 to 15 Increase (Person-rem/ry):	4.58E-03
3 to 15 Increase (Person-rem/ry):	1.05E-02
10 to 15 Delta-LERF:	1.72E-08
3 to 15 Delta-LERF:	3.64E-08
10 to 15 Delta-CCFP:	0.703%
3 to 15 Delta-CCFP:	1.49%
3 to 15 Delta-LERF from Corrosion:	3.64E-08
10 to 15 Delta-LERF from Corrosion:	1.72E-08
Increase in LERF (ILRT 3-to-15 years)	1.62E-08

B8.0 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	0.1663%	0.1663%	0.0166%
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
	0.00008%	0.00158%	0.00004%	0.00170%
	0.00001%	0.00018%	0.00000%	0.00019%
				0.00189%

Release type	FitzPatrick Dose Person-rem	CDF Frequency/ry	Case 3 in 10 yrs Corrosion Addition	Dose Person-rem/ry
1	2.91E+03	6.18E-07		1.80E-03
2	6.79E+05	2.44E-09		1.66E-03
3a	2.91E+04	6.59E-08	4.16E-11	1.92E-03
3b	1.02E+05	6.59E-09	4.62E-12	6.73E-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	6.11E+05	1.74E-07		1.06E-01
7b	3.96E+05	7.37E-07		2.91E-01
7c	5.21E+05	5.82E-08		3.04E-02
7d	3.06E+05	4.83E-07		1.48E-01
8	4.46E+05	2.97E-07		1.32E-01
Total		**2.44E-06**		**0.7146**
			Risk Contribution:	0.3630%
			From 3a and 3b:	2.59E-03
			3b LERF:	6.59E-09
			CCFP:	71.98%

1-in-10 years

Increases to 3a and 3b Frequencies	Drywell/Torus	Inaccessible DW/Torus	Drywell Floor	Total
	0.00034%	0.00686%	0.00017%	0.00737%
	0.00004%	0.00076%	0.00002%	0.00082%
				0.00819%

Release type	FitzPatrick Dose Person-rem	CDF Frequency/ry	Case 1 in 10 yrs	Dose Person-rem/ry
1	2.91E+03	4.49E-07		1.31E-03
2	6.79E+05	2.44E-09	Corrosion Addition	1.66E-03
3a	2.91E+04	2.20E-07	1.80E-10	6.41E-03
3b	1.02E+05	2.20E-08	2.00E-11	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	6.11E+05	1.74E-07		1.06E-01
7b	3.96E+05	7.37E-07		2.91E-01
7c	5.21E+05	5.82E-08		3.04E-02
7d	3.06E+05	4.83E-07		1.48E-01
8	4.46E+05	2.97E-07		1.32E-01
Total		2.44E-06		0.7202
			Risk Contribution:	1.2008%
			From 3a and 3b:	8.65E-03
			3b LERF:	2.20E-08

1-in-15 years

Increases to 3a and 3b Frequencies	Drywell/Torus	Inaccessible DW/Torus	Drywell Floor	Total
	0.00063%	0.01255%	0.00031%	0.00094%
	0.00007%	0.00139%	0.00003%	0.00150%
				0.00244%

Release type	FitzPatrick Dose Person-rem	CDF Frequency/ry	Case 1 in 15 yrs	Dose Person-rem/ry
1	2.91E+03	3.29E-07		9.58E-04
2	6.79E+05	2.44E-09	Corrosion Addition	1.66E-03
3a	2.91E+04	3.29E-07	2.30E-11	9.60E-03
3b	1.02E+05	3.30E-08	3.66E-11	3.36E-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	6.11E+05	1.74E-07		1.06E-01
7b	3.96E+05	7.37E-07		2.91E-01
7c	5.21E+05	5.82E-08		3.04E-02
7d	3.06E+05	4.83E-07		1.48E-01
8	4.46E+05	2.97E-07		1.32E-01
Total		2.44E-06		0.7242
			Risk Contribution:	1.7904%
			From 3a and 3b:	1.30E-02
			3b LERF:	3.30E-08

Other Pertinent Risk Metrics

10 to 15 Increase (Person-rem/ry):	3.97E-03
3 to 15 Increase (Person-rem/ry):	9.53E-03
10 to 15 Delta-LERF:	1.10E-08
3 to 15 Delta-LERF:	2.64E-08
10 to 15 Delta-CCFP:	0.450%
3 to 15 Delta-CCFP:	1.08%
3 to 15 Delta-LERF from Corrosion:	2.64E-08
10 to 15 Delta-LERF from Corrosion:	1.10E-08
Increase in LERF (ILRT 3-to-15 years)	3.06E-10

B9.0 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	4.0703%	4.0703%	0.4070%
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%
	0.00124%	0.00830%	0.00021%	0.00975%
				0.00975%

Release type	FitzPatrick Dose Person-rem	CDF Frequency/ry	Case 3 in 10 yrs	Dose Person-rem/ry
1	2.91E+03	6.18E-07		1.80E-03
2	6.79E+05	2.44E-09	Corrosion Addition	1.66E-03
3a	2.91E+04	6.59E-08	0.00E+00	1.92E-03
3b	1.02E+05	6.83E-09	2.38E-10	6.96E-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	6.11E+05	1.74E-07		1.06E-01
7b	3.96E+05	7.37E-07		2.91E-01
7c	5.21E+05	5.82E-08		3.04E-02
7d	3.06E+05	4.83E-07		1.48E-01
8	4.46E+05	2.97E-07		1.32E-01
Total		2.44E-06		0.7147
			Risk Contribution:	0.3661%
			From 3a and 3b:	2.62E-03
			3b LERF:	6.83E-09
			CCFP:	71.99%

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%
	0.02110%	0.14066%	0.00352%	0.16527%
				0.16527%

Release type	FitzPatrick Dose Person-rem	CDF Frequency/ry	Case 1 in 10 yrs	Dose Person-rem/ry
1	2.91E+03	4.45E-07		1.30E-03
2	6.79E+05	2.44E-09	Corrosion Addition	1.66E-03
3a	2.91E+04	2.20E-07	0.00E+00	6.40E-03
3b	1.02E+05	2.60E-08	4.03E-09	2.65E-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	6.11E+05	1.74E-07		1.06E-01
7b	3.96E+05	7.37E-07		2.91E-01
7c	5.21E+05	5.82E-08		3.04E-02
7d	3.06E+05	4.83E-07		1.48E-01
8	4.46E+05	2.97E-07		1.32E-01
Total		2.44E-06		0.7206
			Risk Contribution:	1.2562%
			From 3a and 3b:	9.05E-03
			3b LERF:	2.60E-08

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%
	0.12252%	0.81681%	0.02042%	0.95975%
				0.95975%

Release type	FitzPatrick Dose Person-rem	CDF Frequency/ry	Case 1 in 15 yrs	Dose Person-rem/ry
1	2.91E+03	3.05E-07		8.90E-04
2	6.79E+05	2.44E-09	Corrosion Addition	1.66E-03
3a	2.91E+04	3.29E-07	0.00E+00	9.60E-03
3b	1.02E+05	5.64E-08	2.34E-08	5.75E-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	6.11E+05	1.74E-07		1.06E-01
7b	3.96E+05	7.37E-07		2.91E-01
7c	5.21E+05	5.82E-08		3.04E-02
7d	3.06E+05	4.83E-07		1.48E-01
8	4.46E+05	2.97E-07		1.32E-01
Total		2.44E-06		0.7265
			Risk Contribution:	2.1129%
			From 3a and 3b:	1.54E-02
			3b LERF:	5.64E-08

Other Pertinent Risk Metrics

10 to 15 Increase (Person-rem/ry):	5.89E-03
3 to 15 Increase (Person-rem/ry):	1.18E-02
10 to 15 Delta-LERF:	3.04E-08
3 to 15 Delta-LERF:	4.95E-08
10 to 15 Delta-CCFP:	1.243%
3 to 15 Delta-CCFP:	2.03%
3 to 15 Delta-LERF from Corrosion:	4.95E-08
10 to 15 Delta-LERF from Corrosion:	3.04E-08
Increase in LERF (ILRT 3-to-15 years)	2.77E-08

ATTACHMENT 5 to JAFP-03-0108

CONTAINMENT INSERVICE INSPECTION PROGRAM SUMMARY

Proposed License Amendment to Provide a One-time Integrated Leak Rate Test (ILRT) Interval Extension

Entergy Nuclear Operations, Inc.
JAMES A. FITZPATRICK NUCLEAR POWER PLANT
Docket No. 50-333
DPR-59

CONTAINMENT DESIGN

James A. Fitzpatrick N.P.P. is a power uprated 881 Mwe, General Electric Boiling Water Reactor (BWR4). The reactor is contained in a Chicago Bridge and Iron Works supplied Mark 1, Free Standing Steel Containment Building. The Mark 1 Containment Building was designed by Stone and Webster Engineering Corporation (Contract No. AP-4).

The design, fabrication, inspection, and testing of the containment conforms to the requirements for Class B vessels in the 1968 Edition of Section III of the ASME Boiler and Pressure Vessel Code for Nuclear Vessels, including the 1968 Summer Addenda.

CONTAINMENT DESCRIPTION

The Drywell is a free standing steel pressure vessel, with a spherical lower portion 65 feet in diameter, and a cylindrical upper portion 35 feet 7 inches in diameter. The overall height of the Drywell is approximately 111 feet 5 ¼ inches. The Drywell is closed at the top by a curved head 30 feet 2 inches in diameter attached at the Drywell neck by a bolted flanged assembly. It is enclosed in reinforced concrete for shielding. There is a sand filled transition zone to accommodate for radial expansion of the spherical section of the freestanding vessel. Above the sand filled transition zone, the Drywell is separated from the reinforced concrete by an air gap of approximately 2 inches. Below the sand filled transition zone the reinforced concrete is in direct contact with the outside surface of the vessel. In addition to the Drywell head, there are two double-door Air Locks and two double-door Equipment Hatches and one Control Rod Drive Removal Hatch to provide access for people and equipment

The suppression chamber (Torus) is a steel pressure vessel in the shape of a torus encircling the Drywell, with a major diameter of approximately 108 feet and a cross-sectional diameter of 29 feet 6 inches. The Torus has three hatch openings, one-48 inch diameter and two-24 inch diameter for personnel and equipment access. The shell is stiffened by 16 internal ring girders located at the miter joints. The Torus is supported by 16 pairs of reinforced W12x161 columns at the ring girder locations. As a result of the Mark I Containment Program, a vent header deflector, saddle plates under the Torus, and anchor bolts with tie-down structures were added. A system of large and small pipes connects the Drywell to the Torus/water. Eight large circular pipes (Vent Pipes), each with a diameter of 6 feet 9 inches, connect the Drywell vessel to the Torus vessel. These eight large circular vent pipes ultimately connect to a circular header vessel (Ring Header) inside the Torus vessel airspace. The eight large circular vent pipes direct steam and vapor following a design bases accident to the Torus ring header. The circular ring header, with a diameter of 4 feet 9 inches directs the steam and vapor through ninety-six, 2 foot diameter pipes, which terminate 4 foot below the Torus normal water level in the Torus vessel. There are eight bellows type expansion joints located inside the Torus vessel that accommodate thermal expansion and vibrational movement between the Drywell vessel and the Torus vessel.

CONTAINMENT INSPECTIONS

A. Background

Structural degradation of containment is a gradual process that occurs due to the effects of pressure, temperature, radiation, chemical or other such effects. Such effects would be identified and corrected when the containment structure is periodically tested and inspected to verify structural integrity.

Effective September, 1996, the NRC endorsed Subsections IWE and IWL of ASME Section XI, 1992 Edition including 1992 Addenda. These subsections contain inservice inspection and repair/replacement rules for Class MC and Class CC components. The reactor containment is a free-standing steel containment, to which only the requirements of Subsection IWE apply.

The First Period (September 28, 1997-March 27, 2001) scheduled IWE examinations were performed in accordance with the requirements of the 1992 Edition,1992 Addenda, and 10CFR50.55a. The Second Period (March 28, 2001 – September 27, 2004) and Third Period (September 28, 2004 – September 27, 2006), are under the requirements of the 1998 Edition (no Addenda) and 10CFR50.55a.

Relief Requests in effect are:

1. TAC No. MA5399 granted relief from the visual inspection of seals and gaskets, Examination Category E-D, Item No. E5.10 and E5.20. A pressure test in accordance with 10CFR50, Appendix J is substituted for these inspections.
2. TAC No. MA9112 granted to continue initial certification and re-certification of ultrasonic testing personnel in accordance with the requirements contained in the 1989 Edition of ASME, Section XI until June 30, 2001.
3. TAC No. MB2946 granted the use of ASME Section XI, Subsection XI, 1998 Edition (no Addenda).

The First Containment Inservice Inspection Period required a General Visual examination be performed to satisfy the requirements stated in 10CFR50.55a (b)(2)(ix)(E). The General Visual examination was performed either directly or remotely by persons qualified in IWE Containment Inspection. The First Containment Inservice Inspection Period satisfied the required expedited examination of containment, outlined in 10CFR50.55a (g)(6)(ii) to be completed prior to September 9, 2001.

General Visual examinations are performed by engineering personnel knowledgeable in the requirements for design, service, inservice inspection , and/or testing of Class MC components.

Detailed Visual examinations are performed by personnel meeting the applicable requirements of IWA-2300 of the 1992 Addenda, in accordance with CP-189, 1991 Edition.

Acceptance of components for continued service is subject to the rules of IWE-3000. Work Requests were used to document the IWE examinations. "Pre-Evaluated IWE Inspection Criteria for RO13 Outage" (Ref. 1) and "ASME Section XI, Subsection IWE, General Visual Containment Inspection Procedure and Pre-Evaluated IWE Inspection Criteria" (Ref. 2) were used to document the IWE method of examination, including acceptance criteria, for the First and Second Periods respectively.

Pre-evaluated Inspection criteria for Ref. (1)

Paint/Coating distress

a. Chipping, chalking, checking, minor pinpoint rust, and minor surface rust are acceptable

b. Flaking/peeling, blistering, major pinpoint rust, undercutting, medium surface rust, and major surface rust require evaluation

Pitting / Gouges / Arc Strikes / Wear / Surface Discontinuities (base metal loss)

a. Minor surface, less than 10% of base metal, is acceptable

b. Medium surface, greater than 10% of base metal, requires evaluation

c. Major surface, greater than 10% of base material, requires evaluation

d. Cracking, requires evaluation

Dents (no signs of base metal stress or paint/coating distress)

a. Minor dent, less than 10% of base metal thickness, is acceptable

b. Major dent, greater than 10% of base metal thickness, requires evaluation

Pre-evaluated Inspection criteria for Ref. (2)

Uncoated surface areas – If any of the relevant conditions listed below are present, further evaluation may be required:

a. Cracking in base metal

b. Discoloration resulting from age, heat, or corrosion

c. Wear exceeding 10% of the design wall thickness

d. Pits, dents, or gouges of the base metal with depth exceeding 10% of the design wall thickness

e. Corrosion which results in discernable base metal loss exceeding 10% of the design wall thickness

f. Discernable bulges

g. Arc strikes

Coated surface areas

a. Any of the conditions listed above for uncoated surfaces

b. Absence of coating

c. Blisters equal to or greater than size No. 6 as specified in ASTM D 714

d. Checking equal to or greater than standard No. 2 as specified in ASTM D 660

e. Cracking equal to or greater than standard No. 6 as specified in ASTM D 661

f. Flaking equal to or greater than standard No. 6 as specified in ASTM D 772

g. Rusting equal to or greater than Rust Grade 7 as specified in ASTM D 610

Bolting assemblies

a. Bending, twisting, stretching or deforming of bolts or studs

b. Missing or loose bolts, studs, nuts, or washers

c. Fractured bolts, studs, or nuts

d. Degradation of protective coatings on bolting surfaces

e. Evidence of coolant leakage near bolting

f. Localized excessive corrosion

g. Misalignment of connection or bolting

Containment supports

a. Any signs of surface irregularities

b. Deformations or structural degradations of fasteners, clamps or other support items, and loss of integrity at bolted or welded connections

c. Missing, detached, or loose support parts and bolting

d. Arc strikes, weld spatter, paint, scoring, roughness, or general corrosion on close tolerance, machined or sliding surfaces

e. Misalignment of supports

f. Improper clearances of guides and stops

g. Wear which visibly reduces the cross-sectional area of the support

h. Abnormal corrosion which reduces the load bearing capacity of the support

i. Cracklike or linear surface flaws

j. Evidence of clamp or non-integral attachment movement, damage, or movement of component insulation due to support movement.

Using industry experience as a guide, evaluations were done of possible suspect surfaces to identify areas which would require augmented examinations (IN 84-01, NRC Generic Letter 87-05, etc.) Currently there are no identified/scheduled augmented inspections.

No conditions existed in the accessible areas which could indicate the presence of, or result in degradation to such inaccessible areas.

B. Drywell

The most recent IWE inspections of the Drywell were performed during RO13, RO14 (First Period), and RO15 (Second Period). The Second Period Drywell inspections will be completed during RO16, October, 2004. These inspections provide a high degree of assurance that any degradation of the containment will be detected and corrected before it may provide a containment leakage path. The inspections to date have not identified degradation that threatens the structural integrity of the containment. All levels of the Drywell were inspected for coating degradation during RO13 (Nov. 1998) and RO14 (Oct. 2000). Areas showing peeling paint or blistering were videotaped. Peeling paint was scraped off of the located areas. In all these areas, evidence of well adhering carbozinc-11 coating was apparent and no further repairs were deemed necessary.

In response to NRC Generic Letter 87-05, it was decided to inspect both the sand cushion drain lines and the bellow seal drain lines. These inspections were to ensure that the drain lines were unplugged and operational as designed.

Three of the four upper sand cushion drain lines were found to be operational . Only one is needed to perform the function that they were designed for, the draining of condensate which may form in the air gap.

Five of the six outer bellows drain lines were verified operational as designed. One line could not be inspected due to the line's inaccessibility. But, because of the redundancy involved in the design of the outer bellows drain lines, it was not deemed necessary to further pursue the inspection of the remaining liner drain.

Visual examination did not indicate present or past entry of moisture in these areas. The sand cushion at JAF is sealed from the air gap by a stainless steel plate and adhesive seal which would have directed any water into the air gap drain lines and prevented collection in the sand cushion.

C. Torus

In 1981 extensive maintenance on Torus coating was done in conjunction with the Torus modifications. The Torus was desludged and inspected in 1989 and 1992. It was inspected by a contracted vendor in April, 1995 and in January, 1997 under the Torus Preservation Program.

In 1996, the NRC endorsed the use of Subsection IWE to ASME Section XI which provided detailed requirements for inservice inspection of containment structure. IWE General Visual and VT-3 examinations of the Torus were done in October/November, 1998 by the ISI Containment engineer and a contracted vendor. This was done at a time when the Torus was drained for the installation of new strainers. Indications found were coating related and did not show degradation of structural integrity. An additional inspection in the area of the most severe coating loss (Bay B) was ultrasonically inspected to determine the actual thickness of the Torus shell plate. Results of the UT inspections found the shell thickness acceptable (readings ranged between 0.620" to 0.690" where the nominal plate thickness is 0.632").

The existing material condition of the Torus is acceptable and does not compromise the structural integrity of the primary pressure boundary. Underwater coating applications were done in April, 1995 and January, 1997. General Visual IWE inspections of the Torus were performed during RO13, RO14, and RO15. The Second Period Drywell inspections will be completed during RO16, October, 2004. The inspection results indicate that no significant corrosion effects have been experienced on the containment vessel. The inspections conducted to date have not identified degradation that threatens the structural integrity of the containment. Various coating issues have been identified, evaluated and recoated as found necessary.

In addition to the IWE requirements, "Suppression Chamber and Drywell Deterioration Inspection, ST-15B, performed as part of overall Primary Containment Leakage Rate Testing Program, was last done during RO14, October 2000, and concluded that the Drywell and Torus interior/exterior surfaces, including Torus supports, showed no evidence of degradation. The Maintenance Rule walk-downs of the interior / exterior surfaces of the primary containment are performed to insure structural integrity for both the Drywell and Torus. These inspections, also have not identified any structural degradation.

D. Pressure retaining bolting

Per the 1998 Edition, pressure retaining bolting only requires General Visual examinations performed in place, Category E-A, per the 1998 Edition. This corresponds to an examination of all bolted connections three times per inspection interval (10 years). Bolted connections need not be disassembled for performance of examinations. But, JAF has committed (TAC NO MB2946, dated 5/1/2002) to perform an additional detailed visual examination on all accessible surface areas of pressure retaining bolted connections, including bolts, studs, nuts, bushings, washers, threads in base material and flange ligaments between fastener holes, using the following guidelines:

1. A detailed visual examination (VT-1) will be performed for areas where flaws or degradations, which exceed ASME Section XI code requirements, are indicated.

2. Pressure retaining bolting indications will be evaluated by the responsible individual for continuous services. If disassembly is required for further evaluation, then a detailed visual examination (VT-1) shall be performed.

3. A detailed visual examination (VT-1) will be performed is a bolted connection is disassembled at the time of a scheduled General Visual examination.

4. A detailed visual examination (VT-1) will be performed when a bolted connection is disassembled at times other than a scheduled visual examination. Procedures are in place to ensure that the integrity of the reassembled bolted connection is maintained.

The bolting examinations are performed during preventive maintenance activities of certain components. These maintenance activities are scheduled to support replacement of the seals and gaskets used in the component connections. Additionally, some of these connections are routinely used during outages, and the examination and testing of these connections is performed to re-establish containment integrity at the end of the outage. Any parts (except seals and gaskets which are exempt) that are replaced are subject to compliance with our Repair/Replacement Program and receive the appropriate inspections at that time.