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W3F1-2014-0052

August 28, 2014

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

SUBJECT: License Amendment Request
Technical Specification Change to Extend the
Type A Test Frequency to 15 Years
Waterford Steam Electric Station, Unit 3
Docket No. 50-382
License No. NPF-38

Dear Sir or Madam:

Pursuant to 10 CFR 50.90, Entergy Operations, Inc. (Entergy) hereby requests the following amendment to Operating License NPF-38 for Waterford Steam Electric Station, Unit 3 (WF3). The proposed change would allow for the extension to the ten-year frequency of the WF3 Type A or Integrated Leak Rate Test (ILRT) that is required by Technical Specification (TS) 6.15 to be extended to 15 years on a permanent basis.

The proposed change has been evaluated in accordance with 10 CFR 50.91(a)(1) using criteria in 10 CFR 50.92(c) and it has been determined that the changes involve no significant hazards consideration. The bases for these determinations are included in the attached submittal.

Similar TS changes were approved for ANO-2 on April 7, 2011 (ML101680380), Palisades on April 6, 2011 (ML110970616), and Nine Mile Point Unit 2 on March 30, 2010 (ML100730032).

The proposed change includes two new commitments. These commitments are summarized in Attachment 7.

Entergy requests approval of the proposed amendment by September 1, 2015. Once approved, the amendment shall be implemented within 30 days.

If you have any questions or require additional information, please contact John Jarrell, Regulatory Assurance Manager, at 504-739-6685.

I declare under penalty of perjury that the foregoing is true and correct.
Executed on August 28, 2014

Sincerely,

MRC/JPJ

Attachments:

1. Analysis of Proposed Technical Specification Change
2. Markups of Technical Specification Pages
3. Clean (Revised) Technical Specification Pages
4. Summary of the Results from Reactor Building Inspections
5. List of Components that Failed Type B or Type C Tests Since 2002
6. Risk Analysis
7. List of Regulatory Commitments

cc: Mr. Marc L. Dapas
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**Attachment 1 to**

**W3F1-2014-0052**

**Analysis of Proposed Technical Specification Change**

## 1.0 DESCRIPTION

This letter is a request to amend Operating License NPF-38 for Waterford Steam Electric Station, Unit 3 (WF3).

The proposed amendment revises WF3 Technical Specification (TS) 6.15, “Containment Leakage Rate Testing Program,” by replacing the reference to Regulatory Guide (RG) 1.163 “Performance-Based Containment Leak-Test Program,” with a reference to Nuclear Energy Institute (NEI) topical report NEI 94-01, Revision 2-A, “Industry Guideline for Implementing Performance−Based Option of 10 CFR Part 50, Appendix J,” as the implementation document used by Entergy Operations, Inc. (Entergy) to develop the WF3 performance-based leakage testing program in accordance with Option B of 10 CFR 50, Appendix J.

NEI 94-01, Revision 2-A describes an approach for implementing the optional performance-based requirements of Option B, including provisions for extending primary containment integrated leak rate test (ILRT) intervals to fifteen (15) years, and incorporates the regulatory positions stated in RG 1.163. In the safety evaluation (SE) issued by NRC letter dated June 25, 2008, the NRC concluded that NEI 94-01, Revision 2, describes an acceptable approach for implementing the optional performance-based requirements of Option B of 10 CFR 50, Appendix J, and found that NEI 94-01, Revision 2, is acceptable for referencing by licensees proposing to amend their TS in regards to containment leakage rate testing, subject to the limitations and conditions noted in Section 4.0 of the SE.

The NRC based its conclusion, in part, on the NEI 94-01, Revision 2-A Executive Summary which notes that the report meets the limitations and conditions of the SE for Revision 2. Revision 2-A of NEI 94-01 was issued in 2008 and included provisions for extending the ILRT interval to fifteen (15) years subject to the limitations and conditions provided in the SE for Revision 2.

In letter dated June 25, 2008, the NRC accepted, with specific limitations, the topical report TR 94-01, Revision 2, and EPRI Report No. 1009325, Revision 2. This acceptance is applicable to nuclear power reactor licensees for which a license was issued under either 10 CFR Part 50 or Part 52 who propose to amend their technical specifications regarding containment leakage rate testing. The final SE defines the basis for the NRC acceptance of the TR. This WF3 submittal will address the conditions and limitations presented in the SE for NEI 94-01 Revision 2.

In accordance with the guidance in NEI 94-01, Revision 2-A, WF3 proposes to extend the interval for the primary containment ILRT, which is currently required to be performed at ten (10) year intervals to no longer than fifteen (15) years from the last ILRT. The current frequency would require the next ILRT to be performed during the fall 2015 refueling outage. The proposed amendment would allow the next ILRT for WF3 to be performed within fifteen (15) years from the last ILRT (i.e., May 21, 2005), as opposed to the current ten (10) year interval. This would allow successive ILRTs to be performed at fifteen (15) year intervals (assuming acceptable performance history).

## 2.0 PROPOSED CHANGE

WF3 TS 6.15, “Containment Leakage Rate Testing Program,” currently states in part,

> A program shall be established to implement the leakage rate testing of the containment as required by 10 CFR 50.54(o) and 10 CFR 50, Appendix J, Option B, as modified by approved exemptions. This program shall be in accordance with the guidelines contained in Regulatory Guide 1.163, “Performance-Based Containment Leak-Test Program,” dated September 1995, except that the next Type A test performed after the May 12, 1991 Type A test shall be performed no later than May 11, 2006.

The proposed change would revise this portion of TS 6.15 by replacing the reference to RG 1.163 with a reference to NEI 94-01, Revision 2-A. The date for the next ILRT is also revised. The changes are underlined.

> A program shall be established to implement the leakage rate testing of the containment as required by 10 CFR 50.54(o) and 10 CFR 50, Appendix J, Option B, as modified by approved exemptions. This program shall be in accordance with the guidelines contained in <u>NEI 94-01, Revision 2-A, “Industry Guideline for Implementing Performance-Based Option of 10 CFR 50, Appendix J,” dated October, 2008</u>, except that the next Type A test performed after the <u>May 21, 2005</u> Type A test shall be performed no later than <u>May 20, 2020</u>.

Attachment 2 contains the existing TS page 6-24 marked-up to show the proposed changes to TS 6.15. Attachment 3 contains a clean (revised) copy of TS page 6-24 including proposed changes to TS 6.15.

## 3.0 BACKGROUND

The testing requirements of 10 CFR 50, Appendix J, provide assurance that leakage from the reactor building, including systems and components that penetrate the reactor building, does not exceed the allowable leakage values specified in the TS, and that periodic surveillance of reactor building penetrations and isolation valves is performed so that proper maintenance and repairs are made during the service life of the reactor building and the systems and components penetrating the reactor building. The limitation on reactor building leakage provides assurance that the reactor building 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: (1) Type A tests, intended to measure the reactor building overall integrated leakage rate; (2) Type B tests, intended to detect local leaks and to measure leakage across pressure-containing or leakage limiting boundaries (other than valves) for reactor building penetrations; and (3) Type C tests, intended to measure reactor building isolation valve leakage. Type B and C tests identify the vast majority of potential reactor building leakage paths. Type A tests identify the overall (integrated) reactor building leakage rate and serve to ensure continued leakage integrity of the reactor building structure by evaluating those structural parts of the reactor building not covered by Type B and C testing.

In 1995, 10 CFR 50, Appendix J, “Primary Reactor Containment Leakage Testing for Water-Cooled Power Reactors,” was amended to provide a performance-based Option B for the

containment leakage testing requirements. Option B requires that test intervals for Type A, Type B, and Type C testing be determined by using a performance-based approach. Performance-based test intervals are based on consideration of the operating history of the component and resulting risk from its failure. The use of the term “performance-based” in 10 CFR 50, Appendix J refers to both the performance history necessary to extend test intervals as well as to the criteria necessary to meet the requirements of Option B.

Also in 1995, RG 1.163 was issued which endorsed NEI 94-01, Revision 0, “Industry Guideline for Implementing Performance-Based Option of 10 CFR 50, Appendix J,” with certain modifications and additions. Option B, in concert with RG 1.163 and NEI 94-01, Revision 0, allows licensees with a satisfactory ILRT performance history (i.e., two consecutive, successful Type A tests) to reduce the test frequency of the containment Type A (ILRT) test from three (3) tests in ten years to one (1) test in ten years. This relaxation was based on an NRC risk assessment contained in NUREG-1493, “Performance-Based Containment Leak-Test Program”, and Electric Power Research Institute (EPRI) TR-104285, “Risk Impact Assessment of Revised Containment Leak Rate Testing Intervals,” both of which illustrated that the risk increase associated with extending the ILRT surveillance interval was very small.

By letter dated December 2, 1996 as supplemented by letters dated February 4 and March 14, 1997, Entergy Operations, Inc. (Entergy) submitted a TS change request concerning the implementation of 10 CFR 50, Appendix J, Option B. In the SE approving this request (letter dated April 10, 1997), the NRC noted the proposed TS changes were in compliance with the requirements of 10 CFR 50, Appendix J, Option B, and are consistent with the guidance in RG 1.163. All of the important elements of the guidance provided in the Staff’s letter to NEI dated November 2, 1995 were included in the proposed WF3 TS.

With the approval of the TS change request, WF3 transitioned to a performance-based ten (10) year frequency for the Type A tests.

Entergy submitted a TS change to extend the ILRT interval from ten years (120 months) to 15 years (180 months) via letter dated July 23, 2001, as supplemented by letters dated September 21, and November 8, 2001. This one-time extension was approved by the NRC in letter dated February 14, 2002.

By letter dated August 31, 2007, NEI submitted Revision 2 of NEI 94-01 and EPRI TR-1009325, Revision 2, “Risk Impact Assessment of Extended Integrated Leak Rate Testing Intervals,” to the NRC Staff for review.

NEI 94-01, Revision 2, describes an approach for implementing the optional performance-based requirements of Option B described in 10 CFR 50, Appendix J, which includes provisions for extending Type A intervals to up to 15 years and incorporates the regulatory positions stated in RG 1.163. It delineates a performance-based approach for determining Type A, Type B, and Type C reactor building leakage rate surveillance testing frequencies. This method uses industry performance data, plant-specific performance data, and risk insights in determining the appropriate testing frequency. NEI 94-01, Revision 2, also discusses the performance factors that licensees must consider in determining test intervals. However, it does not address how to perform the tests because these details are included in existing documents (e.g., American National Standards Institute / American Nuclear Society [ANSI / ANS]-56.8-2002). The NRC final SE issued by letter dated June 25, 2008, documents the NRC’s evaluation and acceptance

of NEI TR 94-01, Revision 2, subject to the specific limitations and conditions listed in Section 4.1 of the SE. The accepted version of NEI 94-01 was subsequently issued as Revision 2-A dated October 2008.

EPRI TR-1009325, Revision 2, provides a risk impact assessment for optimized ILRT intervals of up to fifteen (15) years, utilizing current industry performance data and risk-informed guidance, primarily Revision 1 of RG 1.174, "An Approach for using Probabilistic Risk Assessment in Risk-Informed Decisions on Plant-Specific Changes to the Licensing Bases." The NRC's final SE issued by letter dated June 25, 2008, documents the NRC's evaluation and acceptance of EPRI TR-1009325, Revision 2, subject to the specific limitations and conditions listed in Section 4.2 of the SE. An accepted version of EPRI TR-1009325 was subsequently issued as Revision 2-A (also identified as TR-1018243) dated October 2008.

By letter dated June 25, 2008, the NRC accepted NEI TR 94-01, Revision 2, "Industry Guideline for Implementing Performance−Based Option of 10 CFR Part 50, Appendix J". The NRC staff requested NEI to publish an approved version of NEI TR 94-01, which would incorporate the SE. NEI TR 94-01, Revision 2-A was issued in October 2008 and included guidance for extending the Type A Integrated Leak Rate Testing (ILRT) interval to fifteen (15) years.

## 4.0 TECHNICAL ANALYSIS

As required by 10 CFR 50.54(o), the WF3 reactor building is subject to the requirements set forth in 10 CFR 50, Appendix J. Option B of Appendix J requires that test intervals for Type A, Type B, and Type C testing be determined by using a performance-based approach. Currently, the WF3 10 CFR 50, Appendix J Testing Program Plan is based on RG 1.163, which endorses NEI 94-01, Revision 0. This license amendment request proposes to revise the WF3 10 CFR 50, Appendix J Testing Program Plan by implementing the guidance in NEI 94-01, Revision 2-A.

In the SE issued by the NRC dated June 25, 2008, the NRC concluded that NEI 94-01, Revision 2, describes an acceptable approach for implementing the optional performance-based requirements of Option B of 10 CFR 50, Appendix J, and found that NEI 94-01, Revision 2, is acceptable for referencing by licensees proposing to amend their TS in regards to containment leakage rate testing, subject to the limitations and conditions noted in Section 4.0 of the SE. The following addresses each of the six (6) limitations and conditions associated with Revision 2.

| Limitation / Condition (from Section 4.1 of SE for Revision 2-A) | WF3 Response |
|---|---|
| 1. For calculating the Type A leakage rate, the licensee should use the definition in the NEI TR 94-01, Revision 2, in lieu of that in ANSI/ANS-56.8-2002. | Following the NRC approval of this license amendment request, WF3 will use the definition in Section 5.0 of NEI 94-01, Revision 2-A, for calculating the Type A leakage rate when future WF3 Type A tests are performed (see Attachment 7, "List of Regulatory Commitments"). |
| 2. The licensee submits a schedule of containment inspections to be performed prior to and between Type A tests. | WF3 has a recurring task established to perform a containment building integrity check every three years as well as a recurring task to perform a shield building integrity check to coincide with the performance of the ILRT. A schedule of containment inspections is provided in Section 4.3 below. |
| 3. The licensee addresses the areas of the containment structure potentially subjected to degradation. | General visual examination of accessible interior and exterior surfaces of the containment system for structural problems is typically conducted in accordance with the WF3 Containment In-service Inspection (CISI) Plan which implements the requirements of the ASME, Section XI, Subsections IWE , as required by 10 CFR 50.55a(g). Subsection IWL does not apply to Waterford 3 since the containment vessel does not rely on the detached concrete shell for structural support or pressure retention.<br><br>There are currently no primary containment surface areas that require augmented examinations in accordance with ASME Section XI, IWE-1240. Historical IWE results are discussed in Section 4.4. |
| 4. The licensee addresses any test and inspections performed following major modifications to the containment structure, as applicable. | In December 2012, WF3 replaced the steam generators that required modifications to the containment structure. A snoop (bubble) test of the affected areas of the containment liner was performed with satisfactory results in RF18 in lieu of a Type A test post modification. This was performed per approved request for alternative letter dated July, 27, 2011 |

| Limitation / Condition (from Section 4.1 of SE for Revision 2-A) | WF3 Response |
|---|---|
| | (ML112150195). |
| 5. The normal Type A test interval should be less than 15 years. If a licensee has to utilize the provisions of Section 9.1 of NEI TR 94-01, Revision 2, related to extending the ILRT interval beyond 15 years, the licensee must demonstrate to the NRC staff that it is an unforeseen emergent condition. | Entergy acknowledges and accepts this NRC staff position, as communicated to the nuclear industry in Regulatory Issue Summary (RIS) 2008-27, dated December 8, 2008 (see Attachment 7, "List of Regulatory Commitments"). |
| 6. For plants licensed under 10 CFR Part 52, applications requesting a permanent extension of the ILRT surveillance interval to 15 years should be deferred until after the construction and testing of containments for that design have been completed and applicants have confirmed the applicability of NEI TR 94-01, Revision 2, and EPRI Report No. 1009325, Revision 2, including the use of past containment ILRT data. | Not applicable. WF3 is not licensed pursuant to 10 CFR Part 52. |

To comply with the requirement of 10 CFR 50, Appendix J, Option B, Section V.B, WF3 TS 6.15 currently references RG 1.163. RG 1.163 states that NEI 94-01, Revision 0, provides methods acceptable to the NRC for complying with Option B of 10 CFR 50, Appendix J, with four exceptions described therein.

The current WF3 TS does not list any exceptions to the guidelines contained in RG 1.163.

## 4.1 Previous ILRT Results

NEI 94-01, Revision 2-A, delineates in part (section 8.0, Testing Methodologies for Type A, B, and C tests) that for the purposes of determining an acceptable Type A test for operability considerations, the as-found overall integrated leakage rate shall be calculated by adding the positive difference between the as-found Minimum Path Leakage Rate (MNPLR) and the as-left MNPLR for each pathway tested and adjusted prior to the ILRT (savings); and the as-found MNPLR of all leakage paths isolated during the performance of the ILRT. NEI 94-01, Revision 2-A, further goes on to state in part in Section 8.0 (specifically addressed in the Note) that because of the performance-based emphasis on Type A testing, the performance leakage rate does not use the savings value, but is calculated as the sum of the Type A upper confidence limit and as-left minimum pathway leakage rate for all Type B and Type C pathways that were in service, isolated, or not lined up in their test position. In addition, leakage pathways that were isolated during performance of the test because of excessive leakage must be factored into the

performance determination. The performance criterion for Type A tests is a performance leak rate of less than 1.0 La.

The primary performance-based objective of the Type A test is not to quantify an overall containment system leakage rate, but to ensure continued leakage integrity of the containment structure. Type B and C testing assures that individual penetrations are essentially leak tight. In addition, aggregate Type B and Type C leakage rates support the leakage tightness of primary containment by minimizing potential leakage paths. A review of industry ILRT test results has concluded that almost all containment leakage is identified by local leakage rate testing (type B and C tests). The performance criterion for Type A test allowable leakage is a performance leakage rate of less than 1.0 $L_a$. This allowable performance leakage rate is calculated as the sum of the Type A upper confidence limit and as-left MNPLR for all Type B and Type C pathways that were in service, isolated, or not aligned in the test position prior to performing the Type A test. In addition, leakage pathways that were isolated during performance of the ILRT because of excessive leakage must be factored into the performance determination.

A total of three (3) Type A tests have been performed during the period of time of the WF3 Operating License. A summary of the last two (2) ILRTs for WF3 is provided below.

**May 12, 1991 –** The As-Found Type A test for containment leakage met the 0.75 $L_a$ test requirement criteria as well as all acceptance criteria specified in the test procedure (PE-005-001). The 95% upper confidence limit (UCL) on total time leakage rate, including additions, was 0.0731 wt%/day. The total leakage savings which represents the difference between the as-found and as-left minimum pathway leakage rates from the Type C tests was converted to an equivalent wt%/day and added in the form of a penalty to the as-left ILRT leakage value. Some penetrations were not aligned in the post-accident configuration during the ILRT. The minimum pathway leakage through these penetrations was added to the total time UCL to compensate for the non-conservative valve lineups. Adding the total minimum pathway leakage to the calculated total time 95% UCL of 0.0718 wt%/day yielded the adjusted total time 95% UCL of 0.0731 wt%/day.

It was shown that the results of the leakage rate test would have been acceptable had the test been performed prior to any repairs made during the local leakage rate testing program. This was shown by adding the minimum pathway leakage improvements to the total time 95% UCL. The total minimum pathway improvements summed to 0.0127 wt%/day. This was added to the total time 95% UCL of 0.0731wt%/day resulting in a theoretical as-found leakage rate of 0.0858 wt%/day. This was well below even the as-left acceptance limit of 0.375 wt%/day.

The as-left Type A test for containment leakage utilized the 95% upper confidence limit for leakage, water level corrections, and penalty for valves not being aligned in accident condition. The as-left leakage rate was calculated at 0.0731 wt%/day and 0.0679 wt%/day for Total Time and Mass Point calculational methods, respectively. Therefore containment leakage was less than 0.375 wt%/day and acceptable. ILRT test pressure was at 44.2 psig. In consideration of the performance-based criteria described in NEI 94-01, Revision 2-A, that provides the basis for determining whether historical Type A testing is acceptable, the May 12, 1991, Type A as-left test results of 0.0731 wt%/day and 0.0679wt%/day for Total Time and Mass Point methods, respectively, are acceptable from a Performance-Based approach and ensures leakage integrity of the containment structure.

**May 21, 2005** – Type A test conducted at 45 psig test pressure was successfully performed (design pressure is 47 psig). An acceptable mass point leakage rate at the 95% upper confidence limit of 0.0581 wt%/day was calculated, not including the leakage rate corrections. Total leakage rate corrections (water volume corrections and Type B and C LLRT penalties for valves not aligned in accident condition) were determined to be -0.0039 wt%/day and 0.0025 wt%/day for water level and penalties, respectively. The ILRT occurred at the end of R13 outage. This allowed some penetration boundaries to be repaired or adjusted prior to the ILRT. The total leakage savings (positive difference between the as-found MNPLR and the as-left MNPLR) was documented in STA-001-004, LLRT, Attachment 12.1 and totaled 10,385 sccm or the equivalent of 0.0050 wt%/day. In consideration of the aforementioned inputs, the as-found containment leakage rate was calculated at 0.0617 wt%/day. The acceptance criteria for containment as-found leakage is < La (allowable leakage), or 0.5% wt%/day. Therefore, the as-found ILRT results were acceptable.

The as-left leakage rate was calculated utilizing the 95% upper confidence limit of 0.0581 wt%/day, water level correction of -0.0039 wt%/day, and 0.0025 wt%/day penalty for valves not being aligned in accident alignment. Therefore, the as-left leakage rate was 0.0567 wt%/day. The acceptance criterion for the as-left ILRT is 0.375 wt%/day, so at 0.0567 wt%/day, the leakage results were acceptable.

Based on the above discussion, a test frequency of at least once per fifteen (15) years would be in accordance with NEI 94-01, Revision 2-A.

No modifications that require a Type A test are planned prior to R23, when the next Type A test will be performed under this proposed change. Any unplanned modifications to the reactor building prior to the next scheduled Type A test would be subject to the special testing requirements of Section IV.A of 10 CFR 50, Appendix J. There have been no pressure or temperature excursions in the reactor building which could have adversely affected reactor building integrity. There is no anticipated addition or removal of plant hardware within the reactor building which could affect leak-tightness.

### 4.2 Type B and Type C Testing Program

The WF3 Appendix J, Type B and Type C testing program requires testing of electrical penetrations, airlocks, hatches, flanges, and valves within the scope of the program as required by 10 CFR 50, Appendix J, Option B and TS 6.15. The Type B and Type C testing program consists of local leak rate testing of penetrations with a resilient seal, expansion bellows, double-gasketed manways, hatches and flanges, and containment isolation valves that serve as a barrier to the release of the post-accident containment atmosphere.

A review of the most recent Type B and Type C test results and their comparison with the allowable leakage rate was performed. The combined Type B and Type C leakage acceptance criterion is 630,000 sccm. The maximum and minimum pathway leakage rate summary totals for the last two refueling outages are shown below:

| | |
|---|---|
| RF18 As-Found Minimum Pathway Leakage | 52,520 sccm |
| RF18 As-Left Maximum Pathway Leakage | 48,205 sccm |

| | |
|---|---|
| RF19 As-Found Minimum Pathway Leakage | 56,849 sccm |
| RF19 As-Left Maximum Pathway Leakage | 69,947 sccm |

As discussed in NUREG-1493 and NEI 94-01, Revision 2-A, Type B and Type C tests can identify the vast majority (greater than 95%) of all potential containment leakage paths. This amendment request adopts the guidance in NEI 94-01, Revision 2-A, in place of NEI 94-01, Revision 0, but otherwise does not affect the scope, performance, or scheduling of Type B or Type C tests. Type B and Type C testing will continue to provide a high degree of assurance that containment integrity is maintained.

## 4.3 Supplemental Inspection Requirements

Consistent with the guidance provided in NEI 94-01, Revision 2, Section 9.2.3.2, a general visual examination of accessible interior and exterior surfaces of the containment for structural deterioration that may affect the containment leak-tight integrity is conducted prior to each Type A test and during at least three other outages before the next Type A test if the interval for the Type A test has been extended to 15 years.

Per SE Section 3.1.1.3, “To avoid duplication or deletion of examinations, licensees using NEI TR 94-01, Revision 2, have to develop a schedule for containment inspections that satisfy the provisions of Section 9.2.3.2 of this TR and ASME Code, Section XI, Subsection IWE and IWL requirements.”

These inspections are performed along with the IWE program inspections and are ensured via a recurring task established to perform a containment building integrity check every three years as well as a recurring task to perform a shield building integrity check to coincide with the performance of the ILRT.

Attachment 4 provides a list of the summaries of the results from various WF3 reactor building inspections. It should be noted that Subsection IWL does not apply to Waterford 3 since the containment vessel does not rely on the detached concrete shell for structural support or pressure retention.

Table 4-1 presents summaries of the results from the WF3 containment building interior and exterior structural inspection surveillances. These surveillances were performed every three years and the shield building inspection was performed prior to any integrated leak test.

Table 4-2 presents the IWE inspection summary results. WF3 has three (3) in-service inspection (ISI) periods during each ten (10) year Interval.

The current testing frequencies for Type B and C tests are not affected by this requested amendment to permanently extend the Type A test interval to fifteen (15) years.

## 4.4 Deficiencies Identified

Consistent with the guidance provided in NEI 94-01, Revision 2, Section 9.2.3.3, abnormal degradation of the primary containment structure identified during the conduct of IWE program examinations or at other times is entered into the corrective action program for evaluation to determine the cause of the degradation and to initiate appropriate corrective actions.

Degradation of the Moisture Barrier was previously identified during the first period of the second interval (RF10) and entered into the corrective action program. Inspection of the containment liner below this area of the moisture barrier identified an area of degradation that was added into the IWE program as an augmented examination. In addition to the requirements of Table IWE-2500-1, Owner Elected Examinations were performed every outage from RF10 through RF16. Due to the degradation remaining essentially unchanged over this time period, these areas were evaluated to no longer require augmented examinations in accordance with IWE-2420(c).

Attachment 5 presents a list of components that have failed the Type B or Type C tests from 2005 to present. These conditions were also entered into the corrective action program for evaluation and to initiate appropriate corrective actions.

## 4.5 Plant-Specific Confirmatory Analysis

### 4.5.1 Methodology

An evaluation has been performed to assess the risk impact of extending the WF3 ILRT interval from the current ten (10) years to fifteen (15) years. This plant-specific risk assessment followed the guidance in NEI 94-01, Revision 2-A, the methodology described in EPRI TR-1009325, Revision 2-A and the NRC regulatory guidance outlined in RG 1.174 on the use of Probabilistic Risk Assessment (PRA) findings and risk insights in support of a request to change the licensing basis of the plant. The current WF3 Level 2 and Large Early Release Frequency (LERF) internal events PRA model was used to perform the plant-specific risk assessment. This PRA model has been updated to meet Capability Category II of ASME PRA Standard RA-Sb-2005 and RG 1.200, Revision 1.

The analyses include evaluation for the dominant external events (seismic and fire) using conservative expert judgment with the information from the WF3 Individual Plant Examination of External Events (IPEEE) as well as the Waterford 3 Fire PRA. Though the IPEEE seismic model has not been updated since the original IPEEE, the insights and information of IPEEE have been used to estimate the effect on total LERF of including this external event in the ILRT interval extension risk assessment. Also, the Waterford 3 Fire PRA was used to estimate the effect that this ILRT interval extension will have on the total LERF.

In the SE issued by NRC letter dated June 25, 2008, the NRC concluded that the methodology in EPRI TR-1009325, Revision 2, is acceptable for referencing by licensees proposing to amend their TS to extend the ILRT surveillance interval to fifteen (15) years, subject to the limitations and conditions noted in Section 4.0 of the SE. The following table addresses each of the four limitations and conditions for the use of EPRI TR-1009325, Revision 2.

| Limitation/Condition (From Section 4.2 of SE) | WF3 Response |
|---|---|
| 1. The licensee submits documentation indicating that the technical adequacy of their PRA is consistent with the requirements of RG 1.200 relevant to the ILRT extension application. | WF3 PRA quality is addressed in Section 4.5.2. |
| 2. The licensee submits documentation indicating that the estimated risk increase associated with permanently extending the ILRT surveillance interval to 15 years is small, and consistent with the clarification provided in Section 3.2.4.5 of this SE. Specifically, a small increase in population dose should be defined as an increase in population dose of less than or equal to either 1.0 person-rem per year or 1 percent of the total population dose, whichever is less restrictive. In addition, a small increase in CCFP should be defined as a value marginally greater than that accepted in previous one-time ILRT extension requests. This would require that the increase in CCFP be less than or equal to 1.5 percentage point. While acceptable for this application, the NRC staff is not endorsing these threshold values for other applications. Consistent with this limitation and condition, EPRI Report No. 1009325 will be revised in the "-A" version of the report, to change the population dose acceptance guidelines and the CCFP guidelines. | EPRI Report No. 1009325, Revision 2-A, incorporates these population dose and Conditional Containment Failure Probability (CCFP) acceptance guidelines, and these guidelines have been used for the WF3 plant specific assessment. |
| 3. The methodology in EPRI Report No. 1009325, Revision 2, is acceptable except for the calculation of the increase in expected population dose (per year of reactor operation). In order to make the methodology acceptable, the average leak rate for the pre-existing containment large leak rate accident case (accident case 3b) used by the licensees shall be 100 $L_a$ instead of 35 $L_a$. | EPRI Report No. 1009325, Revision 2-A, incorporated the use of 100 $L_a$ as the average leak rate for the pre-existing containment large leakage rate accident case (accident case 3b), and this value has been used in the WF3 plant specific risk assessment. |
| 4. A licensee amendment request (LAR) is required in instances where containment over-pressure is relied upon for emergency core cooling system (ECCS) performance. | WF3 does not rely on containment overpressure to assure adequate net positive suction head for ECCS pump following design basis accidents |

### 4.5.2 PRA Quality

The WF3 PRA model, Revision 4, combines Level 2 and LERF models for internal events. Severe accident sequences have been developed from internally initiated events. The sequences have been mapped to the radiological release end state (i.e., source term release to environment).

The WF3 PRA is based on a detailed model of the plant developed from the Individual Plant Examination which underwent NRC review. Review comments, current plant design, current procedures, plant operating data, current industry PRA techniques, and general improvements identified by the NRC have been incorporated into the current PRA model. The model is maintained in accordance with Entergy PRA procedures.

The WF3 PRA internal events model has been updated to meet ASME PRA Standard RA-Sb-2005 "Standard for Probabilistic Risk Assessment for Nuclear Power Plant Applications" and RG 1.200, "An Approach for Determining the Technical Adequacy of Probabilistic Risk Assessment Results for Risk-Informed Activities", Revision 1. The industry peer review of the updated PRA model has been performed. The updated PRA model meets ASME Capability Category II requirements by addressing gaps identified by the peer review.

The WF3 Fire PRA (FPRA) model has undergone a Reg. Guide 1.200 Peer Review against Sections 2 and 3 of the ASME PRA Standard. Based on this peer review, the Fire PRA model was found to be consistent with the PRA Standard and able to be used to support risk-informed applications. Continued development of the FPRA model to address findings and observations given in the Peer Review led to revision of the analysis and the need for additional focused scope peer reviews which were performed to address these changes. The Waterford 3 Fire PRA model remains consistent with the ASME PRA Standard and able to support this application.

Based on the conclusions of these reviews, the Waterford 3 internal events and fire PRA models are considered acceptable for use in assessing the risk impact of extending the WF3 reactor building ILRT interval to fifteen (15) years.

### 4.5.3 Summary of Plant-Specific Risk Assessment Results

The findings of the WF3 risk assessment confirm the general findings of previous studies that the risk impact associated with extending the ILRT interval from three in ten years to one in 15 years is small. The WF3 plant-specific results for extending ILRT interval from the current 10 years to 15 years are summarized below.

- The person-rem/year increase in risk contribution from extending the ILRT test frequency from the current ten (10) year interval to a fifteen (15) year interval is 1.20E-2 person-rem/year.
- The risk increase in LERF from extending the ILRT test frequency from the current ten (10) year interval to a fifteen (15) year interval is 1.40E-8/yr.
- The change in conditional containment failure probability (CCFP) from the current ten

(10) year interval to a fifteen (15) year interval is 3.35E-3/yr.

- The change in Type A test frequency from once (1) per ten (10) years to once (1) per fifteen (15) years increases the risk impact on the total integrated plant risk by only 0.005 percent. Also, the change in Type A test frequency from the original three (3) per ten (10) years to once (1) per fifteen (15) years increases the risk only 0.011 percent. Therefore, the risk impact when compared to other severe accident risks is negligible.

- Regulatory Guide 1.174 [6] 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 a once (1) per ten (10) years to once (1) per fifteen (15) years is 1.40E-8/yr. Guidance in Regulatory Guide 1.174 defines very small changes in LERF as below $10^{-7}$/yr, increasing the ILRT interval from ten (10) to fifteen (15) years is therefore considered non-risk significant and the results support this determination. In addition, the change in LERF resulting from a change in the Type A ILRT test interval from a three (3) per ten (10) years to once (1) per fifteen (15) years is 3.36E-8/yr. The delta LERF is also below the guidance classification of a very small change.

- Regulatory Guide 1.174 also encourages the use of risk analysis techniques to help ensure and show that the proposed change is consistent with the defense-in-depth philosophy. Consistency with defense-in-depth philosophy is maintained by demonstrating that the balance is preserved among prevention of core damage, prevention of containment failure, and consequence mitigation. The change in conditional containment failure probability was estimated to be 3.35E-3 (0.36 percent increase) for the proposed change and 8.05E-3 (0.87 percent increase) for the cumulative change of going from a test interval of three (3) in ten (10) years to one (1) in fifteen (15) years. Both CCFP changes meet the criterion of less than 1.5 percent increase obtained from the EPRI guidance document [2]. Therefore the changes in CCFP are considered small and demonstrate that the defense-in-depth philosophy is maintained.

In reviewing these results, the WF3 analysis demonstrates that the change in plant risk is small as a result of this proposed extension of ILRT testing. The change in LERF defined in the analysis for both the baseline and the current cases is within the acceptance criterion.

Details of the WF3 risk assessment are contained in Attachment 6 to this enclosure.

### 4.6 Conclusion

NEI 94-01, Revision 2-A, describes an NRC-accepted approach for implementing the performance-based requirements of 10 CFR 50, Appendix J, Option B. It incorporates the regulatory positions stated in RG 1.163 and includes provisions for extending Type A intervals to fifteen (15) years. NEI 94-01, Revision 2-A delineates a performance-based approach for determining Type A, Type B, and Type C containment leakage rate surveillance test frequencies. Entergy is adopting the guidance of NEI 94-01, Revision 2-A for the WF3 10 CFR Appendix J testing program plan.

Based on the previous ILRT tests conducted at WF3, it may be concluded that extension of the containment ILRT interval from ten (10) to fifteen (15) years represents minimal risk to increased leakage. The risk is minimized by continued Type B and Type C testing performed in accordance with Option B of 10 CFR 50, Appendix J and inspection activities performed as part of the WF3 IWE ISI program.

This experience is supplemented by risk analysis studies, including the WF3 risk analysis provided in Attachment 6. The findings of the WF3 risk assessment confirm the general findings of previous studies, on a plant-specific basis, that extending the ILRT interval from ten (10) to fifteen (15) years results in a very small change to the WF3 risk profile.

## 5.0 REGULATORY ANALYSIS

### 5.1 Applicable Regulatory Requirements/Criteria

The proposed change has been evaluated to determine whether applicable regulations and requirements continue to be met.

10 CFR 50.54(o) requires primary reactor containments for water-cooled power reactors to be subject to the requirements of Appendix J to 10 CFR 50, "Primary Reactor Containment Leakage Testing for Water-Cooled Power Reactors." Appendix J specifies containment leakage testing requirements, including the types required to ensure the leak-tight integrity of the primary reactor containment and systems and components which penetrate the containment. In addition, Appendix J discusses leakage rate acceptance criteria, test methodology, frequency of testing and reporting requirements for each type of test.

RG 1.163 was developed to endorse NEI 94-01, Revision 0 with certain modifications and additions.

The adoption of the Option B performance-based containment leakage rate testing for Type A testing did not alter the basic method by which Appendix J leakage rate testing is performed; however, it did alter the frequency at which Type A, Type B, and Type C containment leakage tests must be performed. Under the performance-based option of 10 CFR 50, Appendix J, the test frequency is based upon an evaluation that reviews "as-found" leakage history to determine the frequency for leakage testing which provides assurance that leakage limits will be maintained. The change to the Type A test frequency did not directly result in an increase in containment leakage. Similarly, the proposed change to the Type A test frequency will not directly result in an increase in containment leakage.

NEI 94-01, Revision 2-A, describes an approach for implementing the performance-based requirements of 10 CFR 50, Appendix J, Option B. The document incorporates the regulatory positions stated in RG 1.163 and includes provisions for extending Type A intervals to fifteen (15) years. NEI 94-01, Revision 2-A, delineates a performance-based approach for determining Type A containment leakage rate test frequencies. In the SE issued by NRC letter dated June 25, 2008, the NRC concluded that NEI 94-01, Revision 2, describes an acceptable approach for implementing the optional performance-based requirements of 10 CFR 50, Appendix J, and is acceptable for referencing by licensees proposing to amend their TS in regards to containment leakage rate testing, subject to the limitations and conditions, noted in Section 4.0 of the SE.

In letter dated June 25, 2008, the NRC requested that NEI publish an approved version of NEI 94-01, Revision 2, incorporating the SE. This version, NEI 94-01, Revision 2-A, was issued in October 2008 and is acceptable for referencing as the implementing document for meeting the performance-based requirements of 10 CFR 50, Appendix J - Option B.

EPRI TR-1009325, Revision 2, provides a risk impact assessment for optimized Integrated Leak Rate Test (ILRT) intervals up to fifteen (15) years, utilizing current industry performance data and risk-informed guidance. NEI 94-01, Revision 2-A, states that a plant-specific risk impact assessment should be performed using the approach and methodology described in TR-1009325, Revision 2, for a proposed extension of the ILRT interval to 15 years. In the

safety evaluation (SE) issued by NRC letter June 25, 2008, the NRC concluded that the methodology in EPRI TR-1009325, Revision 2, is acceptable for referencing by licensees proposing to amend their TS to extend the ILRT surveillance interval to 15 years, subject to the limitations and conditions noted in Section 4.0 of the SE.

Based on the considerations above, (1) there is reasonable assurance that the health and safety of the public will not be endangered by operation in the proposed manner, (2) such activities will continue to be conducted in accordance with the site licensing basis, and (3) the approval of the proposed change will not be inimical to the common defense and security or to the health and safety of the public.

In conclusion, Entergy Operations, Inc. (Entergy) has determined that the proposed change does not require any exemptions or relief from regulatory requirements, other than the TS, and does not affect conformance with any regulatory requirements / criteria.

### 5.2 No Significant Hazards Consideration

A change is proposed to the Waterford Steam Electric Station, Unit 3 (WF3), Technical Specifications 6.15, "Containment Leakage Rate Testing Program." The proposed amendment would replace the reference to Regulatory Guide (RG) 1.163 with a reference to Nuclear Energy Institute (NEI) topical report NEI 94-01, Revision 2-A, dated October, 2008 as the implementation document used by WF3 to develop the WF3 performance-based leakage testing program in accordance with Option B of 10 CFR 50, Appendix J. The proposed amendment would also extend the interval for the primary containment integrated leak rate test (ILRT), which is required to be performed by 10 CFR 50, Appendix J, from ten (10) years to no longer than fifteen (15) years from the last ILRT.

Entergy has evaluated whether or not a significant hazards consideration is involved with the proposed amendment by focusing on the three (3) standards set forth in 10 CFR 50.92, "Issuance of amendment," as discussed below:

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

   Response: No.

   The proposed amendment involves changes to the WF3 Containment Leakage Rate Testing Program. The proposed amendment does not involve a physical change to the plant or a change in the manner in which the plant is operated or controlled. The primary reactor building function is to provide an essentially leak tight barrier against the uncontrolled release of radioactivity to the environment for postulated accidents. As such, the reactor building itself and the testing requirements to periodically demonstrate the integrity of the reactor building exist to ensure the plant's ability to mitigate the consequences of an accident, and do not involve any accident precursors or initiators. Therefore, the probability of occurrence of an accident previously evaluated is not significantly increased by the proposed amendment.

The integrity of the reactor building is subject to two (2) types of failure mechanisms which can be categorized as (1) activity based and (2) time based. Activity based failure mechanisms are defined as degradation due to system and / or component modifications or maintenance. Local leak rate test requirements and administrative controls such as configuration management and procedural requirements for system restoration ensure that the reactor building containment integrity is not degraded by plant modifications or maintenance activities. The design and construction requirements of the reactor building itself combined with the reactor building inspections performed in accordance with ASME, Section XI, the Maintenance Rule and regulatory commitments serve to provide a high degree of assurance that the containment will not degrade in a manner that is detectable only by a Type A test. Based on the above, the proposed amendment does not involve a significant increase in the consequences of an accident previously evaluate.

The proposed amendment adopts the NRC-accepted guidelines of NEI 94-01, Revision 2-A, for development of the WF3 performance-based testing program. Implementation of these guidelines continues to provide adequate assurance that during design basis accidents, the primary containment and its components will limit leakage rates to less than values assumed in the plant safety analyses. The potential consequences of extending the ILRT interval to fifteen (15) years have been evaluated by analyzing the resulting changes in risk. The increase in risk in terms of person-rem per year within fifty (50) miles resulting from design basis accidents was estimated to be acceptably small and determined to be within the guidelines published in RG 1.174. Additionally, the proposed change maintains defense-in-depth by preserving a reasonable balance among prevention of core damage, prevention of containment failure, and consequence mitigation. WF3 has determined that the increase in Conditional Containment Failure Probability due to the proposed change would be very small. Therefore, it is concluded that the proposed amendment does not significantly increase the consequences of an accident previously evaluated.

Based on the above discussion, it is concluded that the proposed change does not involve a significant increase in the probability or consequences of an accident previously evaluated.

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

Response: No.

The proposed amendment adopts the NRC-accepted guidelines of NEI 94-01, Revision 2-A, for the development of the WF3 performance-based leakage testing program, and establishes a fifteen (15) year interval for the performance of the reactor building ILRT. The reactor building and the testing requirements to periodically demonstrate the integrity of the reactor building exist to ensure the plant's ability to mitigate the consequences of an accident, and do not involve any accident precursors or initiators. The proposed change does not involve a physical change to the plant (i.e., no new or different type of equipment will be installed) or a change to the manner in which the plant is operated or controlled.

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

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

   Response: No.

   The proposed amendment adopts the NRC-accepted guidelines of NEI 94-01, Revision 2-A, for the development of the WF3 performance-based leakage testing program, and establishes a fifteen (15) year interval for the performance of the containment ILRT. This amendment does not alter the manner in which safety limits, limiting safety system set points, or limiting conditions for operation are determined. The specific requirements and conditions of the Reactor Building Leakage Rate Testing Program, as defined in the TS, ensure that the degree of the reactor building structural integrity and leak-tightness that is considered in the plant's safety analysis is maintained. The overall reactor building leakage rate limit specified by the TS is maintained, and the Type A, Type B, and Type C containment leakage tests will be performed at the frequencies established in accordance with the NRC-accepted guidelines of NEI 94-01, Revision 2-A.

   Containment inspections performed in accordance with other plant programs serve to provide a high degree of assurance that the containment will not degrade in a manner that is not detectable by an ILRT. A risk assessment using the current WF3 risk model concluded that extending the ILRT test interval from ten (10) years to fifteen (15) years results in a very small change to the WF3 risk profile.

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

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

### 5.3 Environmental Considerations

The proposed amendment does not involve (i) a significant hazards consideration, (ii) a significant change in the types or significant increase in the amounts of any effluent that may be released offsite, or (iii) a significant increase in individual or cumulative occupational radiation exposure. Accordingly, the proposed amendment meets the eligibility criterion for categorical exclusion set forth in 10 CFR 51.22(c)(9). Therefore, pursuant to 10 CFR 51.22(b), no environmental impact statement or environmental assessment need be prepared in connection with the proposed amendment.

## 6.0 PRECEDENCE

This request is similar to TS changes that were approved for ANO-2 on April 7, 2011 (ML101680380), Palisades on April 6, 2011 (ML110970616), and Nine Mile Point Unit 2 on March 30, 2010 (ML100730032).

**Attachment 2 to**

**W3F1-2014-0052**

**Markups of Technical Specification and Technical Specification Bases Pages**

**(1 page)**

## 6.15 CONTAINMENT LEAKAGE RATE TESTING PROGRAM

A program shall be established to implement the leakage rate testing of the containment as required by 10 CFR 50.54(o) and 10 CFR 50, Appendix J, Option B, as modified by approved exemptions. This program shall be in accordance with the guidelines contained in ~~Regulatory Guide 1.163, "Performance-Based Containment Leak-Test Program," dated September 1995~~ NEI 94-01, Revision 2-A, "Industry Guideline for Implementing Performance-Based Option of 10 CFR 50, Appendix J," dated October, 2008, except that the next Type A test performed after the ~~May 12, 1991~~ May 21, 2005 Type A test shall be performed no later than ~~May 11, 2006~~ May 20, 2020.

The peak calculated containment internal pressure for the design basis loss of coolant accident, $P_a$, is 44 psig.

The maximum allowable containment leakage rate, $L_a$, is 0.5% of containment air weight per day at $P_a$

Leakage rate acceptance criteria are:

a. Overall containment leakage rate acceptance criteria is $\leq$ 1.0 $L_a$. During the first unit startup following each test performed in accordance with this program, the overall containment 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 Type A tests.

b. Air lock acceptance criteria are:

    1. Overall air lock leakage rate is $\leq$ 0.05 $L_a$ when tested at $\geq$ $P_a$.

    2. Leakage rate for each door seal is $\leq$ 0.005 $L_a$ when pressurized to $\geq$ 10 psig.

c. Secondary containment bypass leakage rate acceptance criteria is $\leq$ 0.06 $L_a$ when tested at $\geq$ $P_a$.

d. Containment purge valves with resilient seals acceptance criteria is $\leq$ 0.06 $L_a$ when tested at $\geq$ $P_a$.

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

The provisions of Specification 4.0.3 are applicable to the Containment Leakage Rate Testing Program.

**Attachment 3 to**

**W3F1-2014-0052**

**Clean (Revised) Technical Specification Pages**

**(1 page)**

## 6.15 CONTAINMENT LEAKAGE RATE TESTING PROGRAM

A program shall be established to implement the leakage rate testing of the containment as required by 10 CFR 50.54(o) and 10 CFR 50, Appendix J, Option B, as modified by approved exemptions. This program shall be in accordance with the guidelines contained in NEI 94-01, Revision 2-A, "Industry Guideline for Implementing Performance-Based Option of 10 CFR 50, Appendix J," dated October, 2008, except that the next Type A test performed after the May 21, 2005 Type A test shall be performed no later than May 20, 2020.

The peak calculated containment internal pressure for the design basis loss of coolant accident, $P_a$, is 44 psig.

The maximum allowable containment leakage rate, $L_a$, is 0.5% of containment air weight per day at $P_a$.

Leakage rate acceptance criteria are:

a. Overall containment leakage rate acceptance criteria is $\leq 1.0\ L_a$. During the first unit startup following each test performed in accordance with this program, the overall containment 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 Type A tests.

b. Air lock acceptance criteria are:

   1. Overall air lock leakage rate is $\leq 0.05\ L_a$ when tested at $\geq P_a$.

   2. Leakage rate for each door seal is $\leq 0.005\ L_a$ when pressurized to $\geq$ 10 psig.

c. Secondary containment bypass leakage rate acceptance criteria is $\leq 0.06\ L_a$ when tested at $\geq P_a$.

d. Containment purge valves with resilient seals acceptance criteria is $\leq 0.06\ L_a$ when tested at $\geq P_a$.

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

The provisions of Specification 4.0.3 are applicable to the Containment Leakage Rate Testing Program.

**Attachment 4 to**

**W3F1-2014-0052**

**Summary of the Results From**
**Containment Inspections**

## Summary of the Results From Shield Building Inspections

Below is a list of WF3 containment inspections and summaries of the results of those inspections. Note: Subsection IWL does not apply to Waterford 3 since the containment vessel does not rely on the detached concrete shell for structural support or pressure retention.

Table 4-1 presents summaries of the results from the WF3 shield building interior and exterior structural inspection surveillances. These surveillances were performed during each refueling shutdown and prior to any integrated leak test.

Table 4-2 presents the IWE inspection summary results. WF3 has three (3) ISI periods during each ten (10) year Interval.

**Table 4-1**
**Containment Building Interior and Exterior Structural Inspections**

| | |
|---|---|
| May 1988 | The exterior and interior of the shield building were examined prior to pressurization for the first periodic Type A ILRT performed for WF3. No evidence of deterioration was found as documented in letter W3P88-1283 dated August 23, 1988. |
| June 1991 | The exterior and interior surfaces of the shield building were examined prior to pressurization for the second periodic Type A ILRT performed for WF3. No evidence of deterioration was found as documented in letter W3F1-91-0447 dated August 12, 1991. |
| May 2005 | The general visual inspection results reflect compliance with the building structural integrity requirements.<br><br>All accessible areas of the inner shield building were examined from the annulus area. The shield building was inspected from the interior of the annulus. Superficial cracks were found on the surface of the shield building dome. These cracks appeared to be minor in appearance with calcium carbonate deposits on the surface. This was reported during the previous inspection on April 10, 1991. No change was noted at the time of inspection.<br><br>The exterior of the Shield Building was inspected from the +46 elevation up to the Shield Building dome and accessible areas of the exterior Shield Building from the bottom of the cooling towers and RAB (-35 elevation on up). Several small horizontal cracks on multiple vertical Shield Building ‘ribs’ were identified at varying heights. Surface cracks were identified between 3FA & 2FH @ 8A, Westside Dry Cooling Tower, approximately +5 elevation, 20-25 ft in length. Surface cracks were identified between 2FH & 1FH Westside Dry Cooling Tower, approximately +5 elevation, 15-20 ft in length. An examination was performed of the roof of the shield building. Small surface cracking was present as found in the last inspection on 4/30/91. This inspection revealed no apparent growth of cracking, nor any new signs of structural deterioration. |
| November 2009 | The general visual inspection of the containment vessel was performed along with the IWE inspections. Inspection of the shield building noted no changes from the previous inspection. |
| December 2012 | The general visual inspection of the containment vessel was performed along with the IWE inspections and the results were satisfactory. |

**Table 4-2**
**IWE Inspections**

| | |
|---|---|
| November 2003 | An inspection of the containment building integrity was conducted in accordance with Section XI IWE per WO 26901. A general visual inspection was performed including 100% of the accessible containment vessel surface areas and the area around the Fuel Transfer Tube. VT-3 examinations of the coatings on the interior of the containment vessel found areas with flaking, peeling, blistering, and discoloration of the painting (CR-WF3-2003-3082 and CR-WF3-2003-3142). Containment coating is not classified as an IWE component. While IWE does contain separate examination requirements of some non-structural components such as seals, gaskets, and moisture barriers; it does not contain separate examination requirements for containment coating. These locations were repaired and re-inspected in RF12 with satisfactory results.<br><br>VT-3 examinations of the interior moisture barrier (located between the containment vessel and the concrete floor on the ledge on elevation -4) revealed 13 locations where the moisture barrier has failed by a combination of tearing and cracking (CR-WF3-2003-3083). These damaged sections were removed and replaced with new sealant under WO 27307. |
| May 2005 | A general visual inspection of the inside liner plate was performed in accordance with ASME Section XI Subsection IWE. The examination of the liner plate met the screening criteria or was accepted by the responsible Engineer. The general visual inspection results reflect compliance with the building structural integrity requirements.<br><br>All accessible areas of the outer liner plate were examined from the annulus area. The steel liner plate was inspected in all accessible areas and no discrepancies were found. |
| Fall 2006 | Eleven (11) bolted connection inspections were performed in RF14 with satisfactory results. |
| Spring 2008 | The inner and outer moisture barrier sections MB-01 through MB-15 were inspected in RF15 under WO 125996. All sections were satisfactory with the exception of sections MB-02, -03, -05, and -06 which revealed signs of age related degradation and mechanical damage which required repair. |
| November 2009 | The inner and outer moisture barrier sections MB-01 through MB-15 as well as containment surface area inspections of dome quadrants 1 through 9, plates 1 through 162 (with the exception of 71), and the area around the fuel transfer tube were performed in RF16 with satisfactory results. |

| | |
|---|---|
| April 2011 | Twenty-seven (27) program bolted connections were examined in RF17 with satisfactory results. |
| December 2012 | Containment surface area inspections were performed on sections MB-01 through MB-15 in RF18 as well as the moisture barrier inside the annulus from 0° to 138° azimuth. Results of the liner inspections were satisfactory. As a result of the steam generator replacement activities, hydroblasting was performed and water was found standing on the moisture barrier between the 30° and 70° azimuth location. Three 18"x18" moisture barrier sections were removed and the liner examined at the 30°, 42°, and 70° locations to assure no active degradation was present. After replacement of these sections of the moisture barrier, an examination of the repaired moisture barrier areas were performed; the examination results were satisfactory. |
| May 2014 | The inner moisture barrier was inspected in RF19 of items MB-02 through MB-11 with satisfactory results. The outer moisture barrier was inspected in RF18. Twenty-seven (27) program bolted connections were examined in RF19 with satisfactory results. |

**Attachment 5 to**

**W3F1-2014-0052**

**List of Components that Failed Type B or Type C Tests Since 2005**

**List of Components that Failed Type B or Type C Tests Since 2005**

The implementing procedures at Waterford 3 have two limits, an Administrative Limit and an Operation Limit, for component acceptance criteria. Administrative limits are established as "Warning" limits to monitor component leakage performance. The Operation Limit is equivalent to the CEP-APJ-001 Administrative Limit. Below is a list of WF3 components that have failed the operation limit during LLRTs from 2005 to present.

**Table 5-1: LLRT Failures since 2005**

| Pen | Component | Component Description | Date | Outage | Leakage (sccm) |
|---|---|---|---|---|---|
| 60 | FP-602A | FP MVAAA602A – Fire Protection | 4/23/14 | 19 | 630,000 |
| 132 | Bellows | CB EPEN0316 132 – Magnetic Jacks | 4/15/14 | 19 | 270 |
| 67 | HRA-B | Hydrogen Analyzer B | 12/30/12 | 18 | 11,400 |
| 61 | FP-601B | FP MVAAA601B – Fire Protection | 4/22/11 | 17 | 21,260 |
| 10 | CAP-103 | CAP MVAAA103 – Containment Purge – Inlet | 11/07/09 | 16 | 630,000 |
| 10 | CAP-104 | CAP MVAAA104 – Containment Purge – Inlet | 11/07/09 | 16 | 630,000 |
| 11 | CAP-203 | CAP MVAAA203 – Containment Purge – Exhaust | 11/07/09 | 16 | 630,000 |
| 11 | CAP-204 | CAP MVAAA204 – Containment Purge – Exhaust | 11/07/09 | 16 | 630,000 |
| 48 | CAR-202A | CAR MVAAA202A – CARS Exhaust | 11/07/09 | 16 | 630,000 |
| 60 | FP-601A | FP MVAAA601A – Fire Protection | 10/27/09 | 16 | 630,000 |
| 11 | CAP-203 | CAP MVAAA203 – Containment Purge – Exhaust | 05/4/08 | 15 | 630,000 |
| 11 | CAP-204 | CAP MVAAA204 – Containment Purge – Exhaust | 05/4/08 | 15 | 630,000 |
| 12 | CVR-102 | CVR MVAAA102 – Containment Vacuum Relief | 05/12/08 | 15 | 630,000 |
| 43 | BM-109 | BM MVAAA109 – Boron Management, RDT Outlet | 05/16/08 | 15 | 17,000 |
| 48 | CAR-202A | CAR MVAAA202A – | 05/08/08 | 15 | 18,500 |

| Pen | Component | Component Description | Date | Outage | Leakage (sccm) |
|---|---|---|---|---|---|
| | | CARS Exhaust | | | |
| 60 | FP-601A | FP MVAAA601A – Fire Protection | 05/05/08 | 15 | 630,000 |
| 40 | SI-407B | SI MVAAA407B – Shutdown Cooling | 12/06/06 | 14 | 29,000 |
| 49B | ARM-104 | ARM MVAAA104 – Containment Atmosphere Rad Monitor | 11/28/06 | 14 | 630,000 |
| 60 | FP-602A | FP MVAAA602A – Fire Protection | 12/08/06 | 14 | 630,000 |
| 60 | FP-601A | FP MVAAA601A – Fire Protection | 12/18/06 | 14 | 181,000 |
| 13 | CVR-202 | CVR MVAAA202 – Containment Vacuum Relief | 05/13/05 | 13 | 630,000 |
| 24 | CC-713 | CC MVAAA713 – CCW to RCPs and CEDM Cooling Coil | 04/23/05 | 13 | 630,000 |
| 41 | SI-406A | SI MVAAA406A – Shutdown Cooling | 04/29/05 | 13 | 7,200 |
| 41 | SI-407A | SI MVAAA406A – Shutdown Cooling | 04/29/05 | 13 | 7,200 |

**Attachment 6 to**

**W3F1-2014-0052**

**Risk Analysis**

| | | | | | |
|---|---|---|---|---|---|
| ☐ ANO-1 | ☐ ANO-2 | ☐ GGNS | ☐ IP-2 | ☐ IP-3 | ☐ PLP |
| ☐ JAF | ☐ PNPS | ☐ RBS | ☐ VY | ☒ W3 | |
| ☐ NP-GGNS-3 | ☐ NP-RBS-3 | | | | |

| | | |
|---|---|---|
| **CALCULATION COVER PAGE** | **EC #** **52430** |  |
| Design Basis Calc. ☐ YES ☒ NO | ☒ CALCULATION | ☐ EC Markup |
| **Calculation No: ECS14-010** | | **Revision: 0** |
| **Title: Waterford 3 Evaluation of Risk Significance of an ILRT Extension** | | **Editorial** ☐ YES ☒ NO |
| **System(s): MISC** | **Review Org (Department): DEMECH** | |
| **Safety Class:**<br>☐ **Safety / Quality Related**<br>☐ **Augmented Quality Program**<br>☒ **Non-Safety Related** | **Component/Equipment/Structure Type/Number:** | |
| **Document Type: B13.40** | | |
| **Keywords (Description/Topical Codes):**<br><br>**PRA** | | |

**REVIEWS**

| | | |
|---|---|---|
| Name/Signature/Date<br>Vendor Prepared<br>(see page 5) | Name/Signature/Date<br>Howard Brodt<br>See Associated EC | Name/Signature/Date<br>John Russo<br>See Associated EC |
| **Responsible Engineer** | ☐ **Design Verifier**<br>☒ **Reviewer**<br>☒ Comments Attached | **Supervisor/Approval**<br><br>☐ Comments Attached |

| CALCULATION REFERENCE SHEET | CALCULATION NO: ECS14-010<br>REVISION: 0 |
|---|---|

**I. EC Markups Incorporated** (N/A to NP calculations)

1.N/A
2.
3.
4.
5.

| **II. Relationships**: | Sht | Rev | Input Doc | Output Doc | Impact Y/N | Tracking No. |
|---|---|---|---|---|---|---|
| 1. PE-005-001 “Containment Integrated Leak Rate Test” | | 4 | ☒ | ☐ | | |
| 2. PRA-W3-01-001S12 “WF3 Large Early Release Frequency (LERF) Model” | | 1 | ☒ | ☐ | | |
| 3. ECS04-001 | | 0 | ☒ | ☐ | | |
| 4. Waterford 3 Emergency Plan | | 45 | ☒ | ☐ | | |
| 5. | | | ☐ | ☐ | | |

**III. CROSS REFERENCES**:

1. “Industry Guideline for Implementing Performance-Based Option of 10 CFR Part 50, Appendix J,” Revision 3-A, Nuclear Energy Institute, NEI 94-01, July 2012.
2.
3.
4.
5.

**IV. SOFTWARE USED**:

Title: CAFTA Version/Release: 5.4 Disk/CD No.______

**V. DISK/CDS INCLUDED**:

Title:____________________ Version/Release__________ Disk/CD No.______

**VI. OTHER CHANGES**:

| Revision | Record of Revision |
|---|---|
| 0 | Initial issue. |
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# Waterford 3
# Evaluation of Risk Significance of an ILRT Extension

Revision Interim Draft

July 2014

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Entergy

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# Document Revision History

| Revision | Date Released | Principal Author | Reviewer Initials | Approval Initials | Summary of Revision |
|---|---|---|---|---|---|
| Draft | 7/22/2014 | S. Pionke | JM/RS | Not Required | |
| 0 | 8/8/2014 | | RS | RS | Original document |
| 1 | | | | | |
| 2 | | | | | |
| 3 | | | | | |
| 4 | | | | | |
| 5 | | | | | |

| Report Quality Assurance | | |
|---|---|---|
| Attribute<br>(comments are located in comment resolution form or electronically noted in text during review) | Attribute Applicable (Yes/No) | Attribute Reviewed (Yes/No) |
| Title Page or Calculation Cover – Contains the title, client, originator, reviewer and approver. Provides information related to revision, and level of review. | Yes | Yes |
| Review Comment and Resolution Form – This documents the review process and includes the reviewer comments, concurrence and originator resolution. | Yes | Yes |
| Table of Contents, including figures and tables – provides a listing of all major sections, drawings, figures, tables, and illustrations. | Yes | Yes |
| Introduction - summary description of the purpose, scope, and the principle tasks required to meet the project objectives. Analysis boundaries, where applicable function. What is included or excluded from the analysis. | Yes | Yes |
| Methodology - Describe the process and supporting methodology that is sufficient to understand the approach and to support a peer review. Is the method consistent with RSC Engineers and client standards and practices? Does the method document consideration of special issues (e.g., common cause, circular logic, and asymmetry)? | Yes | Yes |
| Analysis and Results - Detailed documentation of the implementation of the task steps that may be supported by report appendices, including any intermediate and final results. Does the analysis use appropriate and verified codes and data input? All figures of event tree and fault trees, and sequence cut sets must be reviewed even if not documented in report. Ensure adequate tables to support assessment such as support systems, success criteria, operator actions, systems addressed in the analysis and dependency. Discussion of system fault tree models, success criteria, application, and system operation as required. Listing and discussion of data selection and application as appropriate. | Yes | Yes |
| Conclusions and Recommendations - A concise presentation of the results of the analysis that answer the objective of the analysis. It should highlight important aspects and findings of the assessment and also provide information related to important assumptions and any conservatism or analysis uncertainty present in the analysis. Recommendations (if any) should be based on analysis results. Limitations of the analysis should be clearly listed. Listing of both general and specific assumptions for system analysis is required. For any quantification adequate truncation requirements should be mentioned. Importance and sensitivity assessments for important contributors and uncertain issues as appropriate. | Yes | Yes |
| List of References – Documents all sources used in the development of the analysis, document, or model that would be necessary to verify or repeat the analysis. References should be included for any non-document files (Visio, Excel, CAFTA, etc.) supporting the report. | Yes | Yes |

<table>
<tr><th colspan="4">Report Quality Assurance</th></tr>
<tr><td colspan="2">Appropriate and Necessary Appendices – Provide adequate supporting documentation to be a able to review and draw conclusions from the report. This would include any applicable appendices such as any “raw” data used in the analysis, any calculations performed to support the analysis that are not documented in a calculation, or appendix containing the analysis cut sets or other results listings as appropriate.</td><td>Yes</td><td>Yes</td></tr>
<tr><td>Reviewer Qualification Statement<br>I certify that I am qualified under the RSC Engineers QA/QC program to perform the review of this document and have examined the above attributes for the most current revision.</td><td colspan="3">Signature/ Date<br>R. Summitt 7/22/14</td></tr>
<tr><td>Approver Qualification Statement<br>I certify that I am a qualified approver under the RSC Engineers QA/QC program.  I have reviewed the completed documentation and the methods, analysis and documentation meet applicable industry practices for concept and conformity.</td><td colspan="3">R. Summitt 8/8/14</td></tr>
</table>

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## 1.0 PURPOSE

The purpose of this report is to provide an estimation of the change in risk associated with extending the Type A integrated leak rate test interval beyond the current 10 years specified by 10 CFR 50, Appendix J, Option B [1] for Waterford Steam Electric Station Unit 3 (WF3). This activity supports a request for an exemption from the performance of the integrated leak rate test (ILRT) during the planned refueling outage number 20. The assessment is consistent with the processes described in the methodology identified in EPRI's guidance document, Risk Impact Assessment of Extended Integrated Leak Rate Testing Intervals [2].

Some of the values calculated in this analysis involve very small changes. The detailed calculations performed to support this report were of a level of mathematical significance necessary to calculate the results recorded [20]. However, the tables and illustrational calculation steps presented may present rounded values to support readability.

### 1.1 SUMMARY OF THE ANALYSIS

The reactor containment leakage test program consists of three tests (Type A, Type B, and Type C) [1]. These tests periodically verify the leak-tight integrity of the primary reactor containment, and the systems (and their components) penetrating the containment. Type A testing is intended to measure the overall integrated leak rate which is the summation of leakage through all potential leakage paths including containment welds, valves, fittings and component which penetrate containment. Type B test measure leakage across each pressure-containing or leakage-limiting boundary for a magnitude of containment penetration seals (i.e. resilient seals, gaskets, sealant compounds, flexible metal seal assemblies, air lock door seals, etc.). The final type of testing, Type C, measures containment isolation valve leakage rates. This type of testing is applicable for any valves that provide a direct connection between the inside and outside atmospheres of the primary reactor containment under normal operation, are required to close automatically upon receipt of a containment isolation signal, are required to operate intermittently under post-accident conditions, and are in main steam and feedwater piping and other system which penetrate containment of direct-cycle boiling water power reactors.

10 CFR 50, Appendix J allows individual plants to extend Type A surveillance testing requirements and to provide for performance-based leak testing. This report documents a risk-based evaluation of the proposed change of the ILRT interval for the WF3. The proposed change would impact testing associated with the current surveillance tests for Type A leakage, procedure PE-005-001 [3]. No change to Type B or Type C testing is proposed at this time.

This analysis utilizes the guidelines set forth in NEI 94-01 [4], the methodology used in the EPRI Report and considers the submittals generated by other utilities.

This calculation evaluates the risk associated with various ILRT intervals as follows:

- 3 years – Interval based on the original requirements of 3 tests per 10 years.
- 10 years – This is the current test interval required for WF3.
- 15 years – Proposed extended test interval.

The analysis utilizes the WF3 PRA results taken from the Level 2 model [5].

The release category and person-rem information is based on the approach suggested by the EPRI guidance document [2].

## 1.2 SUMMARY OF RESULTS/CONCLUSIONS

The specific results are summarized in Table 1 below. Type A testing risk is comprised of EPRI Class 3a and Class 3b. Class 3b is defined as the large early release (LERF) contribution to Type A testing. A breakdown of all the EPRI classifications is contained in Tables 9 and 10 of this report.

Table 1
Summary of Risk Impact on Extending Type A ILRT Test Frequency

| | Risk Impact for 3-years (baseline) | Risk Impact for 10-years (current requirement) | Risk Impact for 15-years |
|---|---|---|---|
| Total integrated risk (person-rem/yr) | 2.60E+2 | 2.60E+2 | 2.60E+2 |
| Type A testing risk (person-rem/yr) | 7.45E-3 | 2.48E-2 | 3.73E-2 |
| % total risk (Type A / total) | 0.003% | 0.010% | 0.014% |
| Type A LERF (Class 3b) (per year) | 8.40E-9 | 2.80E-8 | 4.20E-8 |
| Changes due to extension from 10 years (current) | | | |
| Δ Risk from current (Person-rem/yr) | | | 1.20E-2 |
| % Increase from current (Δ Risk / Total Risk) | | | 0.005% |
| Δ LERF from current (per year) | | | 1.40E-8 |
| Δ CCFP from current | | | 3.35E-3 |
| Changes due to extension from 3 years (baseline) | | | |
| Δ Risk from baseline (Person-rem/yr) | | | 2.87E-2 |
| % Increase from baseline (Δ Risk / Total Risk) | | | 0.011% |
| Δ LERF from baseline (per year) | | | 3.36E-8 |
| Δ CCFP from baseline | | | 8.05E-3 |

The results are discussed below:

- The person-rem/year increase in risk contribution from extending the ILRT test frequency from the current ten (10) year interval to a fifteen (15) year interval is 1.20E-2 person-rem/year.

- The risk increase in LERF from extending the ILRT test frequency from the current ten (10) year interval to a fifteen (15) year interval is 1.40E-8/yr.

- The change in conditional containment failure probability (CCFP) from the current ten (10) year interval to a fifteen (15) year interval is 3.35E-3/yr.

- The change in Type A test frequency from once (1) per ten (10) years to once (1) per fifteen (15) years increases the risk impact on the total integrated plant risk by only 0.005 percent. Also, the change in Type A test frequency from the original three (3) per ten (10) years to once (1) per fifteen (15) years increases the risk only 0.011 percent. Therefore, the risk impact when compared to other severe accident risks is negligible.

- Regulatory Guide 1.174 [6] 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 a once (1) per ten (10) years to once (1) per fifteen (15) years is 1.40E-8/yr. Guidance in Regulatory Guide 1.174 defines very small changes in LERF as below $10^{-7}$/yr, increasing the ILRT interval from ten (10) to fifteen (15) years is therefore considered non-risk significant and the results support this determination. In addition, the change in LERF resulting from a change in the Type A ILRT test interval from a three (3) per ten (10) years to once (1) per fifteen (15) years is 3.36E-8/yr. The delta LERF is also below the guidance classification of a very small change.

- Regulatory Guide 1.174 also encourages the use of risk analysis techniques to help ensure and show that the proposed change is consistent with the defense-in-depth philosophy. Consistency with defense-in-depth philosophy is maintained by demonstrating that the balance is preserved among prevention of core damage, prevention of containment failure, and consequence mitigation. The change in conditional containment failure probability was estimated to be 3.35E-3 (0.36 percent increase) for the proposed change and 8.05E-3 (0.87 percent increase) for the cumulative change of going from a test interval of three (3) in ten (10) years to one (1) in fifteen (15) years. Both CCFP changes meet the criterion of less than 1.5 percent increase obtained from the EPRI guidance document [2]. Therefore the changes in CCFP are considered small and demonstrate that the defense-in-depth philosophy is maintained.

In reviewing these results, the WF3 analysis demonstrates that the change in plant risk is small as a result of this proposed extension of ILRT testing. The change in LERF defined in the analysis for both the baseline and the current cases is within the acceptance criterion.

In addition to the baseline assessment, three sensitivity exercises are included. These analyses are provided in Section 5 and are consistent with the methods outlined in the EPRI guidance document [2].

## 2.0 DESIGN INPUTS

The WF3 PRA is intended to provide “best estimate” results that can be used as input when making risk informed decisions. The PRA provides the most complete results for the WF3 PRA. The inputs for this calculation come from the information documented in the WF3 PRA Level 2 model [5]. Since the WF3 PRA Level 2 model was developed, updates have been performed to the internal events and LERF models, but the update did not perform calculations for the other containment endstates required for the ILRT extension evaluation. Therefore, the Rev. 4 model was deemed to be the most recent documentation that contained all of the necessary inputs to complete the ILRT analysis.

The WF3 release states are summarized in Table 2. WF3 Level 2 results are grouped into four accident sequence states that represent the summation of individual accident categories. The number of sequences comprising each sequence state is also presented in Table 2.

Table 2
Release Category Frequencies

| Release Category | Contributing WF3 Accident Categories | Frequency (/yr) | EPRI Classification |
|---|---|---|---|
| INTACT (S) | 10 | 3.28E-7 | Class 1 |
| LERF[1] | 18 | 5.31E-7 | Class 8 |
| SERF | 9 | 1.76E-9 | Class 6 |
| LATE | 14 | 3.32E-6 | Class 7 |
| Total | N/A | 4.18E-6 | N/A |

1. The LERF contribution for WF3 contains early containment failures due to containment phenomenon and by the EPRI guidance these should be collected in Class 7. To accurately classify the contributions, the LERF contribution is separated to be consistent with the EPRI guidance document [2].

Table 4.3-2 of the WF3 Level 2 model [5] analysis provides the endstate and frequency of the respective endstate. Table 3 shows the classification of each endstate and the totals of each classification. The description of the outcome is used to classify each of the 18 contributing LERF endstates.

Table 3
Decomposition of WF3 LERF Frequency and EPRI Classification

| Endstate | Description of Outcome | Frequency (per year) | EPRI Class |
|---|---|---|---|
| LERF01 | Containment failure following high-pressure (HP) vessel breach (VB) – Non-SBO | 5.12E-9 | 7 |
| LERF02 | Containment failure following HP VB – Non-SBO | ε[1] | 7 |
| LERF03 | Containment failure following low pressure (LP) VB – Non-SBO | 1.56E-9 | 7 |
| LERF04 | Temperature induced (TI) SGTR – Non-SBO | 1.07E-8 | 8 |
| LERF05 | Containment failure following LP VB – Non-SBO | 2.06E-9 | 7 |
| LERF06 | Pressure induced (PI) SGTR – Non-SBO | 2.98E-9 | 8 |
| LERF07 | Containment failure following LP VB – Non-SBO | 3.35E-10 | 7 |
| LERF08 | Loss of isolation – Non-SBO | 1.46E-8 | 2 |
| LERF09 | Containment bypass – Non-SBO | 4.38E-7 | 8 |
| LERF10 | Containment failure following LP VB - SBO | ε[1] | 7 |
| LERF11 | Containment failure following HP VB - SBO | 1.19E-11 | 7 |
| LERF12 | Containment failure following LP VB - SBO | 3.55E-9 | 7 |
| LERF13 | TI-SGTR – SBO | 2.45E-8 | 8 |
| LERF14 | Containment failure following LP VB – SBO | 4.79E-9 | 7 |
| LERF15 | PI-SGTR – SBO | 7.26E-9 | 8 |
| LERF16 | Containment failure following LP VB – SBO | ε[1] | 7 |
| LERF17 | Loss of isolation – SBO | 1.84E-9 | 2 |
| LERF18 | Containment bypass – SBO | 1.41E-8 | 8 |
| | | | |
| Contribution to EPRI Classification 2 | | 1.64E-8 | |
| Contribution to EPRI Classification 7 | | 1.74E-8 | |
| Contribution to EPRI Classification 8 | | 4.98E-7 | |
| Total LERF | | 5.31E-7 | |

1. ε represents a probabilistically insignificant value.

In order to develop the person-rem dose associated to the plant damage state it is necessary to associate each release category with an associated release of radionuclides and from this information to calculate the associated dose.

The EPRI guidance on leak rate testing [2] indicates that a surrogate can be applied and is acceptable for estimating risk and suggests one surrogate source is the results contained in NUREG-1150 [7]. NUREG-1150 examined both pressurized water reactors (PWRs) and boiling water reactors (BWRs). The results presented for BWRs (i.e., Peach Bottom, Grand Gulf) are not considered appropriate for this analysis since the core melt mechanics and design are substantially different between WF3 PWR design and the BWRs. Therefore, their results are excluded from consideration.

NUREG-1150 also analyzed the Zion, Sequoyah, and Surry PWR designs. Sequoyah utilizes an ice condenser design and the presence of ice and restricted flow paths can lead to sequences and conditions that are not found in a large dry containment design such as WF3. Therefore, Sequoyah is not considered a good PWR design for comparison.

Surry is a 3-loop Westinghouse design large dry containment and may be somewhat closer to the WF3 design. However the 3-loop design and power level may influence source term composition. Therefore it is not selected as a surrogate.

The remaining assessed design is Zion. It is a Westinghouse 4-loop design and given the power level and other factors, is considered the best surrogate after examination of the NUREG-1150 analyzed plants.

NUREG/CR-4551 [8] provides the Level 2 analysis and offsite consequence assessment for Zion. Table 4.3-2 of that document provides a summary of consequence results that includes population dose (exposure) within fifty (50) miles for internal events.

The exposure estimates for a range of fifty (50) miles around the Zion site are provided in Table 4 for each reported source term group.

Table 4
Reported Person Rem Estimates for Zion Source Term Groups
(summarized from NUREG/CR-4551)

| Source Term Grouping | Exposure (rem) |
|---|---|
| 1 | 1.69E+05 |
| 2 | 3.76E+05 |
| 31 | 1.93E+05 |
| 33 | 3.66E+04 |
| 61 | 2.76E+05 |
| 64 | 6.06E+05 |
| 65 | 1.40E+06 |
| 66 | 2.90E+05 |
| 67 | 1.35E+06 |
| 68 | 2.72E+06 |
| 69 | 6.93E+05 |
| 70 | 2.18E+06 |
| 71 | 3.91E+06 |
| 72 | 1.56E+06 |
| 100 | 3.38E+06 |
| 101 | 4.42E+06 |
| 103 | 5.80E+06 |
| 104 | 5.46E+06 |
| 105 | 6.49E+06 |
| 106 | 8.47E+06 |
| 107 | 6.27E+06 |
| 136 | 9.00E+06 |

Table 4 (continued)
Reported Person Rem Estimates for Zion Source Term Groups (summarized from NUREG/CR-4551)

| Source Term Grouping | Exposure (rem) |
|---|---|
| 137 | 7.19E+06 |
| 139 | 1.34E+07 |
| 140 | 8.98E+06 |
| 142 | 1.41E+07 |
| 143 | 1.09E+07 |
| 172 | 1.90E+07 |
| 173 | 1.55E+07 |
| 175 | 3.24E+07 |
| 176 | 1.94E+07 |
| 178 | 4.11E+07 |
| 179 | 3.93E+07 |
| 301 | 1.27E+02 |
| 302 | 6.18E+02 |
| 303 | 3.59E+03 |

In order to utilize this information it is necessary to convert it to the form needed in the ILRT analysis. This involves classification into one of the four EPRI classes and then determining the representative person-rem estimates.

Table 3.4-4 in NUREG/CR-4551 [8] provides some guidance with respect to the composition of the source term grouping. The highest contributing release type was credited to the corresponding EPRI class. While multiple release types are contained in Table 3.4-4, only eight of the categories contained the majority of the release. Zion labeled these categories as Is. Leak, SGTR, LS, LL, EL, Alpha, NoCF, and BMT. Class 1 consists of any source term groups that are dominated by no containment failures (NoCF). EPRI Class 2 is related to isolation faults; therefore, source term groups with Is. Leak as the main contributor are placed into this EPRI class. EPRI class 7 is related to early and late phenomena-induced failures. Zion categories LS, LL, EL, Alpha, and BMT are all associated with these types of failures. EPRI Class 8 pertains to containment bypass. The Zion category associated with bypass is SGTR.

For some source term groups, the contributing type of release is not completely dominated by one single category but a mixture of categories all representing the EPRI classes. Occasionally, other contributors (excluding the highest contributor) make up a sizeable portion of the composition. These other contributors occasionally are types of releases that would be classified differently than the highest release contributor. An example is source term group 172, where the highest contributor is Alpha (Class 7), with 52 percent of the release, while the second and third highest are associated with bypass failures (Class 8), combining for 37 percent of the release.. This group was ultimately classified as Class 7 because the Alpha release is considered the more severe type of release and was the highest contributor to the source term group. Using this information the Zion results are grouped to the EPRI classes. The grouping is presented in Table 5.

Table 5
Assignment of Zion Source Term Groups to EPRI Classes

| EPRI Class | Zion Source Term Groups Applied | Average Exposure (person-rem) |
|---|---|---|
| Class 1 | 301, 302 | 7.45E+2 |
| Class 2 | 1, 31, 61, 64, 67, 100 | 5.97E+6 |
| Class 7 | 33, 66, 69, 70, 72, 103, 105, 106, 136, 139, 142, 172, 175, 178, 303 | 1.55E+8 |
| Class 8 | 2, 65, 68, 71, 101, 104,107, 137, 140, 143, 173, 176, 179 | 1.26E+8 |

EPRI's ILRT guidance document [2] utilizes a multiplication factor to develop the design basis leakage value ($L_a$) that is based on generic information that provides comparative local population ratios. The WF3 population dose is adjusted for the local plant-specific population using a "population dose factor". The population dose factor is used to adjust the Zion population dose to account for differences in the local populations of the Zion and WF3 sites. The population dose factor is calculated by dividing the WF3 population [9] by the Zion population information taken from the EPRI ILRT guidance document [2].

Total WF3 Population = 1,998,010

Zion Population = 4,439,288

Population Dose Factor = 0.45

The relationship above implies that the resultant doses are a direct function of population within fifty (50) miles of each site. This does not take into account differences in meteorology, environmental factors, containment designs or other factors but does provide a reasonable first-order approximation of the population dose as would be generated by the Zion accident sequences.

While Zion had two release categories that fell into EPRI Class 1, a more accurate estimate for the INTACT dose rate at WF3 is developed using plant-specific data from Reference 10. The

INTACT dose is the basis for Class 3a and Class 3b doses, which are key in the ILRT delta-dose calculations. Therefore, using plant-specific information to develop the dose associated with INTACT yields results more reflective of the WF3 site.

The method for developing the person-rem dose rate for the population within fifty (50) miles of WF3 utilizes a scaling factor. The dose rates for the exclusion area boundary (EAB) and the low population zone (LPZ) are used to define a distance scaling factor. This scaling factor is then used to estimate the dose for distances beyond the LPZ up to the fifty (50) mile radius.

An average person-rem dose is predicted assuming a uniform distribution of radionuclides that decreases with increased distance from the origin. A uniform distribution of the surrounding population is then combined to calculate the final total dose. The analysis depends on inputs from the licensing basis analysis [10] to arrive at the EAB does rate, LPZ dose rate, LPZ total person-rem dose and population data [9].

The EAB is defined as the circular area within a radius of 914 meters (~0.57 miles) from the containment. The LPZ extends the radius to 3,300 meters (~2.05 miles) from the containment. Table 6 below presents the predicted dose rates for the EAB and LPZ two (2) hours after an event and the thirty (30) day LPZ dose.

Table 6
Predicted Dose from Reference 10

| Location | Dose (rem) |
|---|---|
| $EAB_{2hr}$ | 4.11E+0 |
| $LPZ_{2hr}$ | 6.28E-1 |
| $LPZ_{30d}$ | 2.46E+0 |

The calculation of the necessary scaling factor is based on the relationship of dose rate and distance. The scaling equation is based on a ratio of the LPZ dose to EAB dose. The equation is presented below:

$$Y = X \times \left(\frac{d_{EAB}}{d_{LPZ}}\right)^{C} \quad \text{(eq. 1)}$$

Where:

Y = LPZ dose
X = EAB dose
$d_{LPZ}$ = Distance for LPZ
$d_{EAB}$ = Distance for EAB
C = Scaling Constant

This equation assumes that the dose rate is decreasing in a constant manner with distance and is consistent with the Comanche Peak ILRT submittal [11]. Solving the equation yields a value for the scaling constant (C). The input data is listed below in Table 7.

Table 7
Calculation Parameters Solving for the Scaling Constant (C)

| Parameter | Value (units) |
|---|---|
| X | 4.11E+0 (rem) |
| Y | 6.28E-1 (rem) |
| $d_{EAB}$ | 914 (meters) |
| $d_{LPZ}$ | 3300 (meters) |

Solving Equation 1 with the inputs listed above yields a value of 1.46 for the scaling constant, C. Now the LPZ total dose data can be extrapolated to the fifty (50) mile radius dose criteria.

Equation 1 is utilized again, but instead of solving for the scaling constant the equation is solved for fifty (50) mile radius dose. As the distance from the containment increases the so does the population surrounding the site, but the dose from an event also decreases with distance. Consistent with Comanche Peak ILRT submittal, a value of twenty five (25) miles is used in the extrapolation to represent the average dose for the fifty (50) mile radius since it is the midpoint between the containment and the dose radius parameter. The values displayed in Table 8 are used in the same formula as Equation 1 to solve for the dose at twenty five (25) miles.

Table 8
Calculation Parameters for the Dose at 25 Miles

| Parameter | Value (units) |
|---|---|
| X ($LPZ_{30d}$) | 2.46E+0 (rem) |
| C | 1.46 |
| $d_{LPZ}$ | 2.05 (miles) |
| $d_{25}$ | 25 (miles) |

$$Y = X \times \left(\frac{d_{LPZ}}{d_{25}}\right)^{C} = 2.46 \times \left(\frac{2.05}{25}\right)^{1.46} = 6.33\text{E-}2 \qquad \text{(eq. 2)}$$

Solving Equation 2 with the inputs from Table 8 yields a value for the whole body dose of 6.33E-2 rem. This value represents an average individual dose.

Now that the average person-rem dose rate for the fifty (50) mile radius zone is developed, the effect on the surrounding population is determined. The estimated population is 2.00E+6 persons. However, it is usually assumed that ninety five (95) percent of the population will be evacuated prior to the release such that only five (5) percent of the population would be involved [21]. Given a total population estimate of approximately 2.00E+6 people, this equates to an

exposed population of 9.99E+4 persons. The whole body dose multiplied by the estimated population exposed to a release yields a fifty (50) mile total population whole body dose of 6.33E+3 person-rem.

Table 9 contains the release category dose information. Class 1 dose information is derived from a scaling factor based on plant specific data. Class 2, Class 7, and Class 8 are developed by multiplying the Zion dose for these classes, contained in Table 5, by the population dose factor. Class 6 applies a decontamination factor of 0.1 to the dose associated with Class 2 based on an assumption that 10 percent of the release would be scrubbed.

Table 9
WF3 Dose for EPRI Accident Classes

| Release Category | Frequency (/yr) | EPRI Class | WF3 Dose (person-rem) |
|---|---|---|---|
| INTACT | 3.28E-7 | Class 1 | 6.33E+3 |
| LERF[1] | 1.64E-8 | Class 2 | 2.69E+6 |
| SERF[2] | 1.76E-9 | Class 6 | 2.69E+5[3] |
| LERF + LATE[4] | 3.33E-6 | Class 7 | 6.95E+7 |
| LERF[5] | 4.98E-7 | Class 8 | 5.66E+7 |

1. The EPRI Class 2 category consists of WF3 assigned LERF contribution associated with isolation failures as re-categorized in Table 3.
2. The EPRI Class 6 category consists of WF3 assigned scrubbed isolation failures in SERF.
3. The EPRI Class 6 Does rate is derived from the Class 2 does rate. A decontamination factor of 0.1 is applied with the assumption that 10 percent of the release would be scrubbed.
4. The EPRI Class 7 category consists of the WF3 assigned LERF contribution associated with phenomenological failures as re-categorized in Table 3. Additionally consistent with the EPRI guidance document, LATE failures are classified as Class 7.
5. The EPRI Class 8 category consists of the WF3 assigned LERF contribution associated with bypass or SGTR failures as re-categorized in Table 3.

## 3.0 ASSUMPTIONS

1. The maximum containment leakage for EPRI Class 1 sequences is 1 $L_a$ (Type A acceptable leakage) because a new Class 3 has been added to account for increased leakage due to Type A inspections [2].

2. The maximum containment leakage for Class 3a sequences is 10 $L_a$ based on the EPRI guidance.

3. The maximum containment leakage for Class 3b sequences is 100 $L_a$ based on the NEI guidance contained within the EPRI report.

4. Class 3b is conservatively categorized as LERF based on the NEI guidance and previously approved EPRI methodology.

5. Containment leakage due to EPRI Classes 4 and 5 are considered negligible based on the NEI guidance and the previously approved EPRI methodology.

6. The containment releases are constant and continuous and are not impacted with time. The duration of the release is defined by the LERF definition provided in the PRA.

7. The containment releases for EPRI Classes 2, 6, 7, and 8 are not impacted by the ILRT Type A Test frequency. These classes already include containment failure with release consequences equal or greater than those impacted by Type A. Therefore, the application of a simple WF3 population dose factor to the Zion doses for these classes is considered sufficient.

8. 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.

## 4.0 CALCULATIONS

This calculation applies the WF3 PRA release category information in terms of frequency and person-rem estimates to determine the changes in risk due to increasing the ILRT test interval. The changes in risk are assessed consistent with the guidance provided in the EPRI guidance document [2].

### 4.1 CALCULATIONAL STEPS

The analysis employs the steps provided in EPRI's ILRT guidance document and uses associated risk metrics to evaluate the impact of a proposed change on plant risk. These measures are the change in release frequency, the change in risk as defined by the change in person-rem, the change in LERF, and the change in the conditional containment failure probability (CCFP).

Additionally EPRI also lists the change in CDF as a measure to be considered [2]. Since the testing addresses the ability of the containment to maintain its function, the proposed change has no measurable impact on core damage frequency. Therefore, this attribute remains constant and has no risk significance.

The overall analysis process is documented as outlined below:

- Define and quantify the baseline plant damage classes and person-rem estimates.

- Calculate baseline leakage rates and estimate probability to define the analysis baseline.

- Develop baseline population dose (person-rem) and population dose rate (person-rem/yr).

- Modify Type A leakage estimate to address extension of the Type A test frequency and calculate new population dose rates, LERF and conditional containment failure probability.

- Compare analysis metrics to estimate the impact and significance of the increase related to those metrics.

The first step in the analysis is to define the baseline plant damage classes and person-rem dose measures. Plant damage state information is developed using the WF3 PRA Level 2 PRA results [5]. The containment endstate information and the results of the containment analysis are used to define the representative sequences. The population person-rem dose estimates for the key plant damage classes are based on the application of the method described in the EPRI ILRT guidance document [2].

The product of the person-rem for the plant damage classes and the frequency of the plant damage state is used to estimate the annual person-rem for the particular plant damage state. Summing these estimates produces the annual person-rem dose based on the sequences defined in the WF3 PRA.

The PRA plant damage state definitions considered isolation failures due to Type B and Type C faults and examined containment challenges occurring after core damage and/or reactor vessel failure. These sequences are grouped into key plant damage classes. Using the plant damage state information, bypass, isolation failures and phenomena-related containment failures are identified. Once identified, the sequence was then classified by the EPRI release category definitions. With this information developed, the PRA baseline inputs are completed.

The second step expands the baseline model to address Type A leakage. The PRA did address Type A (liner-related) faults, represented by INTACT accident sequences, and this contribution has been binned into EPRI Class 1. A new estimation using the EPRI methodology must be incorporated to provide a complete baseline. In order to define leakage that can be linked directly to the Type A testing, it is important that only failures that would be identified by Type A testing exclusively be included.

The EPRI ILRT guidance document [2] provides the estimate for the probability of a leakage contribution that could only be identified by Type A testing based on industry experience. This probability is then used to adjust the intact containment category of the WF3 PRA to develop a baseline model including Type A faults.

The release, in terms of person-rem, is developed based on information contained in EPRI's report and is estimated as a leakage increase relative to allowable dose ($L_a$) defined as part of the ILRT.

The predicted probability of Type A leakage is then modified to address the expanded time between testing. This is accomplished by a ratio of the existing testing interval and the proposed test interval. This assumes a constant failure rate and that the failures are randomly dispersed during the interval between the test.

The change due to the expanded interval is calculated and reported in terms of the change in release due to the expanded testing interval, the change in the population person-rem and the change in large early release frequency. The change in the conditional containment failure probability is also developed. From these comparisons, a conclusion is drawn as to the risk significance of the proposed change.

Using this process, the following were performed:

1. Map the WF3 release categories into the 8 release classes defined by the EPRI Report.

2. Calculate the Type A leakage estimate to define the analysis baseline.

3. Calculate the Type A leakage estimate to address the current testing frequency.
4. Modify the Type A leakage estimates to address extension of the Type A test interval.
5. Calculate increase in risk due to extending Type A testing intervals.
6. Estimate the change in LERF due to the Type A testing.
7. Estimate the change in CCFP due to the Type A testing.

## 4.2 SUPPORTING CALCULATIONS

*Step 1: Map the release categories into the 8 release classes defined by the EPRI Report [2]*

EPRI defines eight (8) release classes as presented in Table 10.

Table 10
EPRI Containment Failure Classifications

| EPRI Failure Classification | Description | Interpretation for Assigning WF3 Release Category |
|---|---|---|
| 1 | Containment remains intact with containment initially isolated | Intact containment bins or late basemat attack sequences. |
| 2 | Dependent failure modes or common cause failures | Isolation faults that are related to a loss of power or other isolation failure mode that is not a direct failure of an isolation component |
| 3 | Independent containment isolation failures due to Type A related failures | Isolation failures identified by Type A testing |
| 4 | Independent containment isolation failures due to Type B related failures | Isolation failures identified by Type B testing |
| 5 | Independent containment isolation failures due to Type C related failures | Isolation failures identified by Type C testing |
| 6 | Other penetration failures | Isolation failure with scrubbing or small isolation fails |
| 7 | Induced by severe accident phenomena | Early containment failure sequences as a result of hydrogen burn or other early phenomena |
| 8 | Bypass | Bypass sequence or SGTR |

Table 11 presents the WF3 release category mapping for these eight accident classes. Person-rem per year is the product of the frequency (per year) and the person-rem.

Table 11
WF3 PRA Release Category Grouping to EPRI Classes

| Class | EPRI Description | Frequency | Person-Rem | Person-Rem/yr |
|---|---|---|---|---|
| 1 | Intact containment | 3.28E-7 | 6.33E+3 | 2.08E-3 |
| 2 | Large containment isolation failures | 1.64E-8 | 2.69E+6 | 4.42E-2 |
| 3a | Small isolation failures (liner breach) | To be Determined | | 0.00E+0 |
| 3b | Large isolation failures (liner breach) | To be Determined | | 0.00E+0 |
| 4 | Small isolation failures - failure to seal (type B) | - | | |
| 5 | Small isolation failures - failure to seal (type C) | - | | |
| 6 | Containment isolation failures (dependent failure, personnel errors) | 1.76E-9 | 2.69E+5 | 4.74E-4 |
| 7 | Severe accident phenomena-induced failure (early) | 3.33E-6 | 6.95E+7 | 2.32E+2 |
| 8 | Containment bypass | 4.98E-7 | 5.66E+7 | 2.82E+1 |
| | Total | 4.18E-6 | | 2.60E+2 |

*Step 2: Calculate the Type A leakage estimate to define the analysis baseline (3 year test interval)*

As displayed in Table 11, the WF3 PRA did not identify any release categories specifically associated with EPRI Classes 4 or 5 and the estimate for Class 3 was redistributed back into INTACT. Therefore each of these classes must be evaluated for applicability to this study.

Class 3:

Containment failures in this class are due to leaks such as liner breaches that could only be detected by performing a Type A ILRT. In order to determine the impact of the extended testing interval, the probability of Type A leakage must be calculated.

In order to better assess the range of possible leakage rates, the Class 3 calculation is divided into two classes. Class 3a is defined as a small liner breach and Class 3b is defined as a large liner breach. This division is consistent with the EPRI methodology [2]. The calculation of Class 3a and Class 3b probabilities is presented below.

Calculation of Class 3a Probability

Data presented in the EPRI report [2] contains 2 Type A leakage events out of 217 tests. Using the data a mean estimate for the probability is determined for Class 3a as shown in Equation 3.

$$P_{Class3a} = \frac{n}{N} = \frac{2}{217} = 0.0092 \qquad \text{(eq. 3)}$$

This probability, however, is based on three tests over a ten (10) year period and not the one per ten-year frequency currently employed at WF3 [3]. The probability (0.0092) must be adjusted to reflect this difference and is adjusted in step 3 of this calculation.

Multiplying the CDF times the probability of a Class 3a leak develops the Class 3a frequency contribution in accordance with guidance provided by EPRI. The total CDF includes contributions already binned to LERF. To include these contributions would result in a potentially conservative result. Therefore, the LERF contribution (Class 2 and Class 8) from CDF is removed (1.64E-8/yr and 4.98E-7/yr). The CDF for WF3 is 4.18E-6/yr as presented in Table 11 and is adjusted to remove the LERF contribution.

Therefore the frequency of a Class 3a failure is calculated as:

$FREQ_{class3a}$ = $PROB_{class3a}$ x (CDF– Class 2 – Class 8)

= 0.0092 x (4.18E-6/yr – 1.64E-8/yr – 4.98E-7/yr) = 3.38E-8/yr (eq. 4)

Calculation of Class 3b Probability

To estimate the failure probability given that no failures have occurred, the guidance provided in the EPRI report [2] suggests the use of a non-informative prior. This approach essentially updates a uniform distribution (no bias) with the available evidence (data) to provide a better estimation of an event.

A beta distribution is typically used for the uniform prior with the parameters $\alpha$=0.5 and $\beta$=1. This is then combined with the existing data (no Class 3b events, 217 tests) using Equation 5.

$$p_{Class3b} = \frac{n+\alpha}{N+\beta} = \frac{0+0.5}{217+1} = \frac{0.5}{218} = 0.0023 \qquad \text{(eq. 5)}$$

where: N is the number of tests, n is the number of events (faults) of interest, $\alpha, \beta$ are the parameters of the non-informative prior distribution. From this solution, the frequency for Class 3b is generated using Equation 6 and is adjusted appropriately to address LERF sequences.

$FREQ_{class3b}$ = $PROB_{class3b}$ x (CDF – Class 8 – Class 2)

= 0.0023 x (4.18E-6/yr – 4.98E-7/yr – 1.64E-8/yr) =8.40E-9/yr (eq. 6)

Class 1:

Although the frequency of this class is not directly impacted by Type A testing and the frequency for Class 1 should be reduced by the estimated frequencies in the new Class 3a and Class 3b in order to preserve the total CDF. The revised Class 1 frequency is therefore:

$FREQ_{class1}$ = $FREQ_{class1}$ – ($FREQ_{class3a}$ + $FREQ_{class3b}$) (eq. 7)

$FREQ_{class1}$ = 3.28E-7/yr – (3.38E-8/yr + 8.40E-9/yr) = 2.86E-7/yr

Class 2:

Class 2 represents large containment isolation failures. Class 2 contains contribution to LERF related to isolation failures without scrubbing credited. The frequency of Class 2 is the sum of those release categories identified in Table 3 as Class 2.

$FREQ_{class2}$ = 1.64E-8/yr (eq. 8)

Class 4:

This group consists of all core damage accidents for which a failure-to-seal containment isolation failure of Type B test components occurs. By definition, these failures are dependent on Type B testing, and Type A testing will not impact the probability. Therefore this group is not evaluated further, consistent with the approved methodology.

Class 5:

This group consists of all core damage accidents for which a failure-to-seal containment isolation failure of Type C test components occurs. By definition, these failures are dependent on Type C testing, and Type A testing will not impact the probability. Therefore this group is not evaluated further, consistent with the approved methodology.

Class 6:

The Class 6 group is comprised of isolation faults that occur as a result of the accident sequence progression. For WF3, this class is defined by the WF3 SERF category.

$FREQ_{class6}$ = 1.76E-9/yr (eq. 9)

Class 7:

Class 7 represents early and late containment failure sequences involving phenomena related containment breach. Class 7 contains contributions to LERF related to early release phenomena. The frequency of Class 7 is the sum of those release categories identified in Table 3 as Class 7 and the frequency associated with LATE failures.

$FREQ_{class7}$ = 3.33E-6/yr (eq. 10)

Class 8:

The frequency of Class 8 is the sum of those release categories identified in Table 3 as Class 8.

$FREQ_{class8}$ = 4.98E-7/yr (eq. 11)

Table 12 summarizes the above information by the EPRI defined classes. This table also presents dose exposures previously calculated. Class 3a and 3b person-rem values are developed based on the design basis assessment of the intact containment as defined in the EPRI guidance report [2].

The Class 3a and 3b doses are represented as $10L_a$ and $100L_a$ respectively. Table 12 also presents the person-rem frequency data determined by multiplying the failure class frequency by the corresponding exposure.

Table 12
Baseline Risk Profile

| Class | Description | Frequency (/yr) | Person-rem | Person-rem (/yr) |
|---|---|---|---|---|
| 1 | No Containment Failure | 2.86E-7 | 6.33E+3 | 1.81E-3 |
| 2 | Large Containment Isolation Failures | 1.64E-8 | 2.69E+6 | 4.42E-2 |
| 3a | Small Isolation Failures (Liner breach) | 3.38E-8 | 6.33E+4[2] | 2.14E-3 |
| 3b | Large Isolation Failures (Liner breach) | 8.40E-9 | 6.33E+5[3] | 5.31E-3 |
| 4 | Small isolation failures - failure to seal (type B) | ε[1] | | |
| 5 | Small isolation failures - failure to seal (type C) | ε[1] | | |
| 6 | Containment Isolation Failures (dependent failure, personnel errors) | 1.76E-9 | 2.69E+5 | 4.74E-4 |
| 7 | Severe Accident Phenomena-induced Failure (Early and Late) | 3.33E-6 | 6.95E+7 | 2.32E+2 |
| 8 | Containment Bypass | 4.98E-7 | 5.66E+7 | 2.82E+1 |
| | Total | 4.18E-6 | | 2.60E+2 |

1. ε represents a probabilistically insignificant value.
2. 10 times $L_a$.
3. 100 times $L_a$.

The percent risk contribution due to Type A testing is defined as follows:

$\%Risk_{BASE} = [( Class3a_{BASE} + Class3b_{BASE}) \mathbf{/} Total_{BASE}] \times 100$ (eq. 12)

Where:

$Class3a_{BASE}$ = Class 3a person-rem/yr for baseline interval = 2.14E-3 person-rem/yr

$Class3b_{BASE}$ = Class 3b person-rem/yr for baseline interval = 5.31E-3 person-rem/yr

$Total_{BASE}$ = total person-rem/yr for baseline interval = 2.60E+2 person-rem/yr

$\%Risk_{BASE}$ = [(2.14E-3 + 5.31E-3) / 2.60E+2] x 100 = **0.003 percent** (eq. 13)

*Step 3: Calculate the Type A leakage estimate to address the current inspection interval*

The current surveillance testing requirement for Type A testing and allowed by 10 CFR 50, Appendix J is at least once (1) per ten (10) years based on an acceptable performance history (defined as two consecutive periodic Type A tests at least twenty four (24) months apart in which the calculated performance leakage was less than $1.0L_a$).

According to the ERRI report [2], extending the Type A ILRT interval from three (3) in ten (10) years to one (1) in ten (10) years will increase the average time that a leak detectable only by an ILRT goes undetected from eighteen (18) to sixty (60) months. Multiplying the testing interval by 0.5 and multiplying by twelve (12) to convert from “years” to “months” calculates the average time for an undetected condition to exist.

The increase for a ten (10) year ILRT interval is the ratio of the average time for a failure to detect for the increased ILRT test interval (from eighteen (18) months to sixty (60) months) multiplied by the existing Class 3a probability as shown in Equation 14.

$$P_{Class3a}(10yr) = 0.0092 \times \frac{60}{18} = 0.0307 \qquad \text{(eq. 14)}$$

A similar calculation is performed for the Class 3b probability as presented in Equation 15.

$$P_{Class3a}(10yr) = 0.0023 \times \frac{60}{18} = 0.0077 \qquad \text{(eq. 15)}$$

<u>Risk Impact due to ten (10) year Test Interval</u>

Based on the approved EPRI methodology [2] and the NEI guidance [4], the increased probability of not detecting excessive leakage due to Type A tests directly impacts the frequency of the Class 3 sequences.

The risk contribution is determined by multiplying the Class 3 accident frequency by the increase in the probability of leakage. Additionally the Class 1 frequency is adjusted to maintain the overall core damage frequency constant. The results of this calculation are presented in Table 13 below.

Table 13
Risk Profile for Once in Ten Year Testing

| Class | Description | Frequency (/yr) | Person-rem [2] | Person-rem (/yr) |
|---|---|---|---|---|
| 1 | No Containment Failure[1] | 1.87E-7 | 6.33E+3 | 1.19E-3 |
| 2 | Large Containment Isolation Failures | 1.64E-8 | 2.69E+6 | 4.42E-2 |
| 3a | Small Isolation Failures (Liner breach) | 1.13E-7 | 6.33E+4 | 7.12E-3 |
| 3b | Large Isolation Failures (Liner breach) | 2.80E-8 | 6.33E+5 | 1.77E-2 |
| 4 | Small isolation failures - failure to seal (type B) | ε[3] | | |
| 5 | Small isolation failures - failure to seal (type C) | ε[3] | | |
| 6 | Containment Isolation Failures (dependent failure, personnel errors) | 1.76E-9 | 2.69E+5 | 4.74E-4 |
| 7 | Severe Accident Phenomena-induced Failure (Early and Late) | 3.33E-6 | 6.95E+7 | 2.32E+2 |
| 8 | Containment Bypass | 4.98E-7 | 5.66E+7 | 2.82E+1 |
| | Total | 4.18E-6 | | 2.60E+2 |

1. The PRA frequency of Class 1 has been reduced by the frequency of Class 3a and Class 3b in order to preserve total CDF.
2. From Table 12.
3. ε represents a probabilistically insignificant value.

Using the same methods as for the baseline, and the data in Table 13 the percent risk contribution due to Type A testing is as follows:

$\%Risk_{10} = [(Class3a_{10} + Class3b_{10})$ **/** $Total_{10}] \times 100$ (eq. 16)

Where:

$Class3a_{10}$ = Class 3a person-rem/yr for current 10-year interval = 7.12E-3 person-rem/yr

$Class3b_{10}$ = Class 3b person-rem/yr for current 10-year interval = 1.77E-2 person-rem/yr

$Total_{10}$ = total person-rem/yr for current 10-year interval = 2.60E+2 person-rem/yr

$\%Risk_{10}$ = [(7.12E-3 + 1.77E-2) / 2.60E+2] x 100 = **0.01 percent** (eq. 17)

The percent risk increase ($\Delta\%Risk_{10}$) due to a ten (10) year ILRT over the baseline case is as follows:

$\Delta\%Risk_{10}$ = [((Class$1_{10}$ + Class$3a_{10}$-+ Class$3b_{10}$) – (Class$1_{BASE}$ + Class$3a_{BASE}$ + Class$3b_{BASE}$))/ Total$_{BASE}$] x 100.0 (eq. 18)

Where:

Class$1_{10}$ = Class 1 person-rem/yr for current 10-year interval = 1.19E-3 person-rem/yr

Class$3a_{10}$ = Class 3a person-rem/yr for current 10-year interval = 7.12E-3 person-rem/yr

Class$3b_{10}$ = Class 3b person-rem/yr for current 10-year interval = 1.77E-2 person-rem/yr

Class$1_{BASE}$ = Class 1 person-rem/yr for baseline interval = 1.81E-3 person-rem/yr (Table 12)

Class$3a_{BASE}$ = Class 3a person-rem/yr for baseline interval = 2.14E-3 person-rem/yr (Table 12)

Class$3b_{BASE}$ = Class 3b person-rem/yr for baseline interval = 5.31E-3 person-rem/yr (Table 12)

Total$_{BASE}$ = total person-rem/yr for baseline interval = 2.60E+2 person-rem/yr (Table 12)

$\Delta\%Risk_{10}$ = [(1.19E-3 + 7.12E-3 + 1.77E-2) – (1.81E-3 + 2.14E-3 + 5.31E-3)] / 2.60E+2 x 100.0 = **0.006 percent** (eq. 19)

*Step 4: Calculate the Type A leakage estimate to address extended inspection intervals*

If the test interval is extended to one (1) per fifteen (15) years, the average time that a leak detectable only by an ILRT test goes undetected increases to ninety (90) months (0.5 x 15 x 12). For a fifteen (15) year test interval, the result is the ratio (90/18) of the exposure times as was the case for the 10 year case. Increasing the ILRT test interval from once (1) every three (3) years to once (1) per fifteen (15) years results in a proportional increase in the overall probability of leakage.

The approach for developing the risk contribution for a fifteen (15) year interval is the same as that for the ten (10) year interval. The increase for a fifteen (15) year ILRT interval is the ratio of the average time for a failure to detect for the increased ILRT test interval (from eighteen (18) months to ninety (90) months) multiplied by the existing Class 3a probability as shown in Equation 20.

$$P_{Class3a}(10yr) = 0.0092 \times \frac{90}{18} = 0.0461 \quad \text{(eq. 20)}$$

A similar calculation is performed for the Class 3b probability as presented in Equation 21.

$$P_{Class3a}(10yr) = 0.0023 \times \frac{90}{18} = 0.0115 \quad \text{(eq. 21)}$$

Risk Impact due to 15-year Test Interval

As stated for the ten (10) year case, the increased probability of not detecting excessive leakage due to Type A tests directly impacts the frequency of the Class 3 sequences.

The increased risk contribution is determined by multiplying the Class 3 accident frequency by the increase in the probability of leakage. Additionally the Class 1 frequency is adjusted to maintain the overall core damage frequency constant. The results of this calculation are presented in Table 14 below.

Table 14
Risk Profile for Once in Fifteen Year Testing

| Class | Description | Frequency (/yr) | Person-rem [2] | Person-rem (/yr) |
|---|---|---|---|---|
| 1 | No Containment Failure[1] | 1.17E-7 | 6.33E+3 | 7.42E-4 |
| 2 | Large Containment Isolation Failures | 1.64E-8 | 2.69E+6 | 4.42E-2 |
| 3a | Small Isolation Failures (Liner breach) | 1.69E-7 | 6.33E+4 | 1.07E-2 |
| 3b | Large Isolation Failures (Liner breach) | 4.20E-8 | 6.33E+5 | 2.66E-2 |
| 4 | Small isolation failures - failure to seal (type B) | ε[3] | | |
| 5 | Small isolation failures - failure to seal (type C) | ε[3] | | |
| 6 | Containment Isolation Failures (dependent failure, personnel errors) | 1.76E-9 | 2.69E+5 | 4.74E-4 |
| 7 | Severe Accident Phenomena-induced Failure (Early and Late) | 3.33E-6 | 6.95E+7 | 2.32E+2 |
| 8 | Containment Bypass | 4.98E-7 | 5.66E+7 | 2.82E+1 |
| | Total | 4.18E-6 | | 2.60E+2 |

1. The PRA frequency of Class 1 has been reduced by the frequency of Class 3a and Class 3b in order to preserve total CDF.
2. From Table 12.
3. ε represents a probabilistically insignificant value.

Using the same methods as for the baseline, and the data in Table 14 the percent risk contribution due to Type A testing is as follows:

$$\%Risk_{15} = [( Class3a_{15} + Class3b_{15}) \,/\, Total_{15}] \times 100 \qquad \text{(eq. 22)}$$

Where:

$Class3a_{15}$ = Class 3a person-rem/yr for 15-year interval = 1.07E-2 person-rem/yr

$Class3b_{15}$ = Class 3b person-rem/yr for 15-year interval = 2.66E-2 person-rem/yr

$Total_{15}$ = total person-rem year for 15-year interval = 2.60E+2 person-rem/yr

$\%Risk_{15}$ = [(1.07E-2 + 2.66E-2) / 2.60E+2] x 100 = **0.014 percent** (eq. 23)

The percent risk increase ($\Delta\%Risk_{15}$) due to a fifteen-year ILRT over the baseline case is as follows:

$\Delta\%Risk_{15}$ = [(($Class1_{15}$ + $Class3a_{15}$ + $Class3b_{15}$) – ($Class1_{BASE}$ + $Class3a_{BASE}$ + $Class3b_{BASE}$))/ $Total_{BASE}$] x 100.0 (eq. 24)

Where:

$Class1_{15}$ = Class 1 person-rem/yr for current 15-year interval = 7.42E-4 person-rem/yr

$Class3a_{15}$ = Class 3a person-rem/yr for current 15-year interval = 1.07E-2 person-rem/yr

$Class3b_{15}$ = Class 3b person-rem/yr for current 15-year interval = 2.66E-2 person-rem/yr

$Class1_{BASE}$ = Class 1 person-rem/yr for baseline interval = 1.81E-3 person-rem/yr (Table 12)

$Class3a_{BASE}$ = Class 3a person-rem/yr for baseline interval = 2.14E-3 person-rem/yr (Table 12)

$Class3b_{BASE}$ = Class 3b person-rem/yr for baseline interval = 5.31E-3 person-rem/yr (Table 12)

$Total_{BASE}$ = total person-rem/yr for baseline interval = 2.60E+2 person-rem/yr (Table 12)

$\Delta\%Risk_{15}$ = [(7.42E-4 + 1.07E-2 + 2.66E-2) – (1.81E-3 + 2.14E-3 + 5.31E-3)] / 2.60E+2 x 100.0 = **0.011 percent** (eq. 25)

*Step 5: Calculate increase in risk due to extending Type A inspection intervals*

Based on the guidance in the EPRI guidance document [2], the percent increase in the total integrated plant risk from a fifteen-year ILRT over a current ten-year ILRT is computed as follows:

$\%Total_{10\text{-}15}$ = [(($Class1_{15}$ + $Class3a_{15}$ + $Class3b_{15}$) – ($Class1_{10}$ + $Class3a_{10}$ + $Class3b_{10}$))/ $Total_{10}$] x 100.0 (eq. 26)

Where:

$Class1_{15}$ = Class 1 person-rem/yr for current 15-year interval = 7.42E-4 person-rem/yr

$Class3a_{15}$ = Class 3a person-rem/yr for current 15-year interval = 1.07E-2 person-rem/yr

$Class3b_{15}$ = Class 3b person-rem/yr for current 15-year interval = 2.66E-2 person-rem/yr

$Class1_{10}$ = Class 1 person-rem/yr for current 10-year interval = 1.19E-3 person-rem/yr (Table 13)

$Class3a_{10}$ = Class 3a person-rem/yr for current 10-year interval = 7.12E-3 person-rem/yr (Table 13)

$Class3b_{10}$ = Class 3b person-rem/yr for current 10-year interval = 1.77E-2 person-rem/yr (Table 13)

$Total_{10}$ = total person-rem/yr for 10-year interval = 2.60E+2 person-rem/yr (Table 13)

% $Total_{10\text{-}15}$ = [(7.42E-4 + 1.07E-2 + 2.66E-2) – (1.19E-3 + 7.12E-3 + 1.77E-2)] / 2.60E+2 x 100 = **0.005 percent** (eq. 27)

*Step 6: Calculate the change in Risk in terms of Large Early Release Frequency (LERF)*

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 larger release due to failure to detect a pre-existing leak during the relaxation period.

From the EPRI Report, the Class 3a dose is assumed to be ten (10) times the intact containment leakage, $L_a$ (or 6.33E+4 person-rem) and the Class 3b dose is assumed to be 100 times $L_a$ (or 6.33E+5 person-rem). The method for defining the dose equivalent for allowable leakage ($L_a$) is developed in the EPRI report. This compares to a historical observed average of twice $L_a$. Therefore, the estimate is somewhat conservative.

Based on EPRI guidance, only 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 $2L_a$). 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.

Therefore, the change in the frequency of Class 3b sequences is used as the increase in LERF for WF3, and the change in LERF can be determined by the differences. The EPRI guidance document [2] identifies that Class 3b is considered to be the main contributor to LERF. Table 15 summarizes the results of the LERF evaluation that Class 3b is indicative of a LERF sequence.

Table 15
Impact on LERF due to Extended Type A Testing Intervals

| ILRT Inspection Interval | 3 Years (baseline) | 10 Years | 15 Years |
|---|---|---|---|
| Class 3b (Type A LERF) | 8.40E-9/yr | 2.80E-8/yr | 4.20E-8/yr |
| ΔLERF (3 year baseline) | | 1.96E-8/yr | 3.36E-8/yr |
| ΔLERF (10 year baseline) | | | 1.40E-8/yr |

Regulatory Guide 1.174 [6] provides guidance for determining the risk impact of plant-specific changes to the licensing basis. The EPRI report [2] cites Regulatory Guide 1.174 and defines very small changes in risk as resulting in increases of core damage frequency (CDF) below 1E-6/yr and increases in LERF below 1E-7/yr. Since the ILRT does not impact CDF, the relevant

metric is LERF. Calculating the increase in LERF requires determining the impact of the ILRT interval on the leakage probability.

By increasing the ILRT interval from the currently acceptable ten (10) years to a period of fifteen (15) years results in an increase in the contribution to LERF of 1.40E-8/yr. This value meets the guidance in Regulatory Guide 1.174 defining very small changes in LERF. The LERF increase measured from the original three (3) in ten (10) year interval to the fifteen (15) year interval is 3.36E-8/yr, which is also less than the criterion presented in Regulatory Guide 1.174.

*Step 7: Calculate the change in Conditional Containment Failure Probability (CCFP)*

The conditional containment failure probability (CCFP) is defined as the probability of containment failure given the occurrence of an accident. This probability can be expressed using the following equation:

$$CCFP = 1 - \left[\frac{f(ncf)}{CDF}\right] \qquad \text{(eq. 28)}$$

Where, *f(ncf)* is the frequency of those sequences which result in no containment failure. This frequency is determined by summing the Class 1 and Class 3a results, and CDF is the total frequency of all core damage sequences.

Therefore the change in CCFP for this analysis is the CCFP using the results for fifteen (15) years ($CCFP_{15}$) minus the CCFP using the results for ten (10) years ($CCFP_{10}$). This can be expressed by the following:

$$\Delta CCFP_{10-15} = CCFP_{15} - CCFP_{10} \qquad \text{(eq. 29)}$$

Using the data previously developed the change in CCFP from the current testing interval is calculated and presented in Table 16.

Table 16
Impact on Conditional Containment Failure Probability due to Extended Type A Testing Intervals

| ILRT Inspection Interval | 3 Years (baseline) | 10 Years | 15 Years |
|---|---|---|---|
| *f*(ncf) (/yr) | 3.20E-7 | 3.00E-7 | 2.86E-7 |
| *f*(ncf)/CDF | 7.65E-2 | 7.18E-2 | 6.85E-2 |
| CCFP | 9.23E-1 | 9.28E-1 | 9.32E-1 |
| ΔCCFP (3 year baseline) | | 4.69E-3 | 8.05E-3 |
| ΔCCFP (10 year baseline) | | | 3.35E-3 |

The EPRI guidance document [2] provides insight for determining acceptable levels of increase in CCFP. The guidance states that an increase in CCFP less than 1.5 percent is considered small based on past ILRT submittals accepted by the NRC.

By increasing the ILRT interval from the currently acceptable ten (10) years to a period of fifteen (15) years results in a CCFP increase of 3.35E-3 or 0.36 percent. This value meets the guidance contained in the EPRI report for small changes in CCFP. The CCFP increase measured from the original three (3) in ten (10) year interval to the fifteen (15) year interval is 8.05E-3 or 0.87 percent, which is also less than the criterion presented in the guidance document.

## 5.0 SENSITIVITY STUDIES

This section presents sensitivity studies suggested in the EPRI report [2] for the WF3 ILRT extension assessment. This includes an evaluation of assumptions made in relation to liner corrosion, the use of the expert elicitation, and the impact of external events.

### 5.1 LINER CORROSION

The analysis approach utilizes the Calvert Cliffs Nuclear Plant (CCNP) methodology [19] as modified by EPRI. This methodology is an acceptable approach to incorporate the liner corrosion issue into the integrated leak rate test (ILRT) extension risk evaluation, but more instances of corrosion have occurred since the EPRI report was published. Therefore the methodology used by CCNP and EPRI will remain unchanged, but the inputs will are updated using a data collection period that begins in September of 1996 and ends on December 31$^{st}$ 2013. Thus the data collection period is extended from 5.5 years to 17.25 years.

Over the 17.25 years, more containment liner corrosion events occurred. In 2011, the NRC published a technical letter report that analyzed containment liner corrosion events occurring at operating nuclear power plants in the USA [12]. The results of this analysis were five (5) containment liner corrosion events in almost fifteen (15) years at sixty six (66) possible sites in the Unites States. Two (2) of the five (5) events are the same existences of corrosion used by CCNP in their liner corrosion analysis (North Anna Power Station Unit 2 and Brunswick Steam Electric Plant Unit 2). The next event took place at D.C. Cook Unit 2 in March of 2001. A small hole was discovered in the liner plate that the plant suspected was man made. In 2009, a through-wall penetration caused by a piece of wood embedded in the concrete was identified at Beaver Valley. It should be noted that in 2006 during the Beaver Valley Unit 1 steam generator replacement surface corrosion was identified. This corrosion had yet to cause penetration in the liner, but since the discovery of this corrosion occurred during a steam generator replacement and not a normal inspection, the event will be included with the conservative assumption that the corrosion would have been discovered after it penetrated the steel liner. The last event occurred in the fall of 2013 at Beaver valley Unit 1 [13]. Thus over the 17.25 year data collection period six (6) liner corrosion events occurred at a possible sixty six (66) plant locations.

Table 17 summarizes the results obtained from the CCNP methodology utilizing a more recent data collection period.

Table 17
WF3 Liner Corrosion Risk Assessment Results Using CCNP Methodology

| Step | Description | Containment Cylinder and Dome (85%) | | Containment Basemat (15%) | |
|---|---|---|---|---|---|
| 1 | Historical liner flaw likelihood<br>Failure data: containment location specific<br>Success data: based on 70 steel-lined containments and 5.5 years since the 10CFR 50.55a requirements of periodic visual inspections of containment surfaces | Events 6<br>6 / (66 x 17.25) = 5.27E-3/yr | | Events: 0<br>Assume a half failure<br>0.5 / (66 x 17.25) = 4.39E-4/yr | |
| 2 | Aged adjusted liner flaw likelihood<br>During the 15-year interval, assume failure rate doubles every five years (14.9% increase per year). The average for the 5th to 10th year set to the historical failure rate. | Year<br>1<br>average 5-10<br>15 | Failure rate<br>2.14E-3/yr<br>5.27E-3/yr<br>1.49E-2/yr | Year<br>1<br>average 5-10<br>15 | Failure rate<br>1.78E-4/yr<br>4.39E-4/yr<br>1.24E-3/yr |
| | | 15 year average = 6.42E-3/yr | | 15 year average = 5.58E-4/yr | |
| 3 | Increase in flaw likelihood between 3 and 15 years<br>Uses aged adjusted liner flaw likelihood (Step 2), assuming failure rate doubles every five years. | 0.74% (1 to 3 years)<br>4.24% (1 to 10 years)<br>9.63% (1 to 15 years) | | 0.06% (1 to 3 years)<br>0.36% (1 to 10 years)<br>0.84% (1 to 15 years) | |
| 4 | Likelihood of breach in containment given liner flaw | 1% | | 0.1% | |

Table 17 (continued)
WF3 Liner Corrosion Risk Assessment Results Using CCNP Methodology

| Step | Description | Containment Cylinder and Dome (85%) | Containment Basemat (15%) |
|---|---|---|---|
| 5 | Visual inspection detection failure likelihood | 10%<br>5% failure to identify visual flaws plus 5% likelihood that the flaw is not visible (not through-cylinder but could be detected by ILRT)<br>All events have been detected through visual inspection. 5% visible failure detection is a conservative assumption. | 100%<br>Cannot be visually inspected |
| 6 | Likelihood of non-detected containment leakage (Steps 3 x 4 x 5) | 0.00074% (3 years)<br>0.74% x 1% x 10%<br>0.00424% (10 years)<br>4.24% x 1% x 10%<br>0.00963% (15 years)<br>9.63% x 1% x 10% | 0.00006% (3 years)<br>0.06% x 0.1% x 100%<br>0.00036% (10 years)<br>0.36% x 0.1% x 100%<br>0.00084% (15 years)<br>0.84% x 0.1% x 100% |

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

Total likelihood of non-detected containment leakage (3 yr) = 0.00074% + 0.00006% = 0.0008%

Total likelihood of non-detected containment leakage (10 yr) = 0.00424% + 0.00036% = 0.0046%

Total likelihood of non-detected containment leakage (15 yr) = 0.00963% + 0.00084% = 0.01047%

This likelihood is then multiplied by the non-LERF containment failures for WF3. This value is calculated by the following equation for each period of interest. LERF is comprised of Class 2, Class 8, and Class 3b cases as shown below in Equation 30.

Non-LERF = CDF – Class 2 – Class 8 – Class 3b (eq. 30)

A final adjustment could be made to address cases with successful containment spray operation. It is conservatively not addressed as it would not be expected to substantially alter the overall results. Table 18 presents the data and the resultant increase in LERF due to liner corrosion for each case.

Table 18
Liner Corrosion LERF Adjustment Using CCNP Methodology

| Case | CDF (/yr) | Class 2 (/yr) | Class 8 (/yr) | Class 3b (/yr) | Likelihood of Non-detected Corrosion Leakage | Increase in LERF (/yr) |
|---|---|---|---|---|---|---|
| 3-years | 4.18E-6 | 1.64E-8 | 4.98E-7 | 8.40E-9 | 8.00E-6 | 2.92E-11 |
| 10-years | 4.18E-6 | 1.64E-8 | 4.98E-7 | 2.80E-8 | 4.60E-5 | 1.67E-10 |
| 15-years | 4.18E-6 | 1.64E-8 | 4.98E-7 | 4.20E-8 | 1.05E-4 | 3.79E-10 |

This contribution is added to the Class 3b LERF cases and the sensitivity analysis performed. Table 19 provides a summary of the base case as well as the corrosion sensitivity case. The “Delta Person-Rem” column provides the change in person-rem between the case without corrosion and the case that considers corrosion. Values within parentheses “( )” indicate the LERF change or delta between the without corrosion and corrosion cases.

Table 19
WF3 Summary of Base Case and Corrosion Sensitivity Cases

| EPRI Class | Base Case (3 per 10 years) | | | | | 1 per 10 years | | | | | 1 per 15 years | | | | |
|---|---|---|---|---|---|---|---|---|---|---|---|---|---|---|---|
| | Without Corrosion | | With Corrosion | | | Without Corrosion | | With Corrosion | | | Without Corrosion | | With Corrosion | | |
| | Frequency | Person-rem per year | Frequency | Person-rem per year | Delta Person-Rem per year | Frequency | Person-rem per year | Frequency | Person-rem per year | Delta Person-Rem per year | Frequency | Person-rem per year | Frequency | Person-rem per year | Delta Person-Rem per year |
| 1 | 2.86E-7 | 1.81E-3 | 2.86E-7 | 1.81E-3 | -1.85E-7 | 1.87E-7 | 1.19E-3 | 1.87E-7 | 1.19E-3 | -1.06E-6 | 1.17E-7 | 7.42E-4 | 1.17E-7 | 7.44E-4 | -2.40E-6 |
| 2 | 1.64E-8 | 4.42E-2 | 1.64E-8 | 4.42E-2 | N/A | 1.64E-8 | 4.42E-2 | 1.64E-8 | 4.42E-2 | N/A | 1.64E-8 | 4.42E-2 | 1.64E-8 | 4.42E-2 | N/A |
| 3a | 3.38E-8 | 2.14E-3 | 3.38E-8 | 2.14E-3 | N/A | 1.13E-7 | 7.12E-3 | 1.13E-7 | 7.12E-3 | N/A | 1.69E-7 | 1.07E-2 | 1.69E-7 | 1.07E-2 | N/A |
| 3b | 8.40E-9 | 5.31E-3 | 8.43E-9 | 5.33E-3 | 1.85E-5 | 2.80E-8 | 1.77E-2 | 2.82E-8 | 1.78E-2 | 1.06E-4 | 4.20E-8 | 2.66E-2 | 4.24E-8 | 2.68E-2 | 2.40E-4 |
| 6 | 1.76E-9 | 4.74E-4 | 1.76E-9 | 4.74E-4 | N/A | 1.76E-9 | 4.74E-4 | 1.76E-9 | 4.74E-4 | N/A | 1.76E-9 | 4.74E-4 | 1.76E-9 | 4.74E-4 | N/A |
| 7 | 3.33E-6 | 2.32E+2 | 3.33E-6 | 2.32E+2 | N/A | 3.33E-6 | 2.32E+2 | 3.33E-6 | 2.32E+2 | N/A | 3.33E-6 | 2.32E+2 | 3.33E-6 | 2.32E+2 | N/A |
| 8 | 4.98E-7 | 2.82E+1 | 4.98E-7 | 2.82E+1 | N/A | 4.98E-7 | 2.82E+1 | 4.98E-7 | 2.82E+1 | N/A | 4.98E-7 | 2.82E+1 | 4.98E-7 | 2.82E+1 | N/A |
| CDF | 4.18E-6 | 2.60E+2 | 4.18E-6 | 2.60E+2 | 1.83E-5 | 4.18E-6 | 2.60E+2 | 4.18E-6 | 2.60E+2 | 1.05E-4 | 4.18E-6 | 2.60E+2 | 4.18E-6 | 2.60E+2 | 2.37E-4 |
| Class 3b LERF | 8.40E-9 | | 8.43E-9 (2.92E-11) | | | 2.80E-8 | | 2.82E-8 (1.67E-10) | | | 4.20E-8 | | 4.24E-8 (3.79E-10) | | |
| Delta LERF (from base case of 3 per 10 years) | | | | | | 1.96E-8 | | 1.97E-8 (1.38E-10) | | | 3.36E-8 | | 3.39E-8 (3.50E-10) | | |
| Delta LERF from 1 per 10 years | | | | | | N/A | | | | | 1.40E-8 | | 1.42E-8 (2.12E-10) | | |

The inclusion of corrosion does not result in an increase in LERF sufficient to invalidate the baseline analysis and the overall impact of corrosion inclusion is negligible.

## 5.2 DEFECT SENSITIVITY AND EXPERT ELICIATION SENSITIVITY

A second sensitivity case on the impacts of assumptions regarding pre-existing containment defect or flaw probabilities of occurrence and magnitude, or size of the flaw, is performed as described in the EPRI guidance document [2]. The expert elicitation contained in the EPRI report developed probabilities for pre-existing containment defects that would be detected by the ILRT only based on the historical testing data.

Using the expert knowledge, this information was extrapolated into a probability versus magnitude relationship for pre-existing containment defects. The failure mechanism analysis also used the historical ILRT data augmented with expert judgment to develop the results. Details of the expert elicitation process and results are contained in the EPRI report. The expert elicitation process has the advantage of considering the available data for small leakage events, which have occurred in the data, and extrapolates those events and probabilities of occurrence to the potential for large magnitude leakage events.

The expert elicitation results are used to develop sensitivity cases for the risk impact assessment. Employing the results requires the application of the ILRT interval methodology using the expert elicitation to change in the probability of pre-existing leakage in the containment.

The baseline assessment uses the Jefferys non-informative prior and the expert elicitation sensitivity study uses the results of the expert elicitation. In addition, given the relationship between leakage magnitude and probability, larger leakage that is more representative of large early release frequency, can be reflected. For the purposes of this sensitivity, the same leakage magnitudes that are used in the basic methodology (i.e., 10 La for small and 100 La for large) are used here. Table 20 presents the magnitudes and probabilities associated with the Jefferys non-informative prior and the expert elicitation use in the base methodology and this sensitivity case.

Table 20
Comparison of Jefferys Non-Informative Prior and Expert Elicitation Values

| Leakage Size ($L_a$) | Jefferys Non-Informative Prior | Expert Elicitation Mean Probability of Occurrence | Percent Reduction |
|---|---|---|---|
| 10 | 9.20E-3 | 3.88E-03 | 58% |
| 100 | 2.30E-3 | 2.47E-04 | 89% |

Taking the baseline analysis and using the values provided in Table 20 for the expert elicitation, the results in Table 21 are developed.

Table 21
WF3 Summary of ILRT Extension Using Expert Elicitation Values

| Accident Class | ILRT Interval | | | | | | | |
|---|---|---|---|---|---|---|---|---|
| | 3 per 10 Years | | | | 1 per 10 years | | 1 per 15 Years | |
| | Base Frequency | Adjusted Base Frequency | Dose (person-rem) | Dose Rate (person-rem/yr) | Frequency | Dose Rate (person-rem/yr) | Frequency | Dose Rate (person-rem/yr) |
| 1 | 3.28E-7 | 3.13E-7 | 6.33E+3 | 1.98E-3 | 2.77E-7 | 1.76E-3 | 2.52E-7 | 1.60E-3 |
| 2 | 1.64E-8 | 1.64E-8 | 2.69E+6 | 4.42E-2 | 1.64E-8 | 4.42E-2 | 1.64E-8 | 4.42E-2 |
| 3a | N/A | 1.43E-8 | 6.33E+4 | 9.03E-4 | 4.76E-8 | 3.01E-3 | 7.14E-8 | 4.52E-3 |
| 3b | N/A | 9.09E-10 | 6.33E+5 | 5.75E-4 | 3.03E-9 | 1.92E-3 | 4.54E-9 | 2.87E-3 |
| 6 | 1.76E-9 | 1.76E-9 | 2.69E+5 | 4.74E-4 | 1.76E-9 | 4.74E-4 | 1.76E-9 | 4.74E-4 |
| 7 | 3.33E-6 | 3.33E-6 | 6.95E+7 | 2.32E+2 | 3.33E-6 | 2.32E+2 | 3.33E-6 | 2.32E+2 |
| 8 | 4.98E-7 | 4.98E-7 | 5.66E+7 | 2.82E+1 | 4.98E-7 | 2.82E+1 | 4.98E-7 | 2.82E+1 |
| Totals | 4.18E-6 | 4.18E-6 | 1.30E+8 | 2.60E+2 | 4.18E-6 | 2.60E+2 | 4.18E-6 | 2.60E+2 |
| Δ LERF (3 per 10 yrs base) | | | | | 2.12E-9 | | 3.63E-9 | |
| Δ LERF (1 per 10 yrs base) | | | | | | | 1.51E-9 | |
| CCFP | 92.17% | | | | 92.22% | | 92.25% | |

The results illustrate how the expert elicitation reduces the overall change in LERF and the overall results are more favorable with regard to the change in risk.

## 5.3 POTENTIAL IMPACTS FROM EXTERNAL EVENTS

An assessment of the impact of external events is performed. The primary basis for this investigation is the determination of the total LERF following an increase in the ILRT testing interval from three (3) in ten (10) years to one (1) in fifteen (15) years.

External events were evaluated in the WF3 Individual Plant Examination of External Events (IPEEE) [14]. The IPEEE program was a one-time review of external hazard risk and was limited in its purpose to the identification of potential plant vulnerabilities and an understanding of severe accident risk. The primary areas of external event analysis for the WF3 IPEEE were seismic and internal fires, and other external events. Seismic and fire were considered to be the most limiting due to their frequency of occurrence and their potential impact on plant operability. Therefore it is assumed that they bound the risk contribution from other external events. Both seismic and internal fire were examined but the analysis contained conservative

assumptions related to consequential failures due to external events such that the absolute CDF is considered an understatement of plant performance and an over estimation of CDF.

The WF3 site is a very low seismicity site and the potential for a seismic event of significance is very low relative to more active locations. Seismic events were addressed through a Seismic Margin Analysis (SMA) as part of the IPEEE for WF3. The Seismic PRA method screened all the components that met a high confidence low probability of failure (HCLPF) for the review level seismic event occurring with a magnitude of 0.3g. The remaining components were grouped together as a proxy component. It was assumed that if this proxy component failed it would result in core damage. This method is considered conservative.

The SMA information is used in conjunction with the improvements that have been incorporated into the internal event model since the IPEEE was performed. Prior seismic analyses have indicated that for a well-designed plant, seismic contributions are a combination of low acceleration events with random failures and higher acceleration events with dependent component or structural failures due to forces associated with the seismic event.

As cited in NUREG-1742 [15], the controlling failure typically involves prolonged loss of ac power leading to a station blackout. Low acceleration events lead to a disruption of offsite power sources and result in a prolonged need for onsite sources. This contribution has been estimated utilizing the current internal events analysis and based on the loss of offsite power (LOSP) initiating events analysis to define a conditional core damage probability (CCDP). This value is then combined with a typical estimation for the median capacity of the offsite power supply (0.3g, median capacity) [22]. The frequency is multiplied by 0.5 for the likelihood of failure of offsite sources given a seismic event.

The CCDP is calculated by modifying the WF3 CAFTA model [16] to only calculate the CCDP associated with loss of offsite power scenarios. The model contains seven (7) unique initiating events (IEs) that are associated with LOSP. Since the impact of any of the seven (7) initiating events is the same, only one event (%T5) is set to a value of 1.0 to represent a condition reflecting a loss of offsite power and the quantification yields the CCDP due to LOSP. The quantification assumes that offsite power cannot be restored within twenty four (24) hours. Since the standard recovery techniques utilize non-seismic data, it is not applicable. The calculated CCDP for SBO without recovery is 1.35E-2. From the seismic hazard curve [17], a 0.3g seismic event has a median frequency of 1.20E-5/yr. At this seismicity level, the best estimate fragility for loss of power yields a probability of 0.5. Combining the frequency, the CCDP and the probability of losing offsite power yields an estimate for the frequency contribution for low acceleration seismic events. The seismic CDF estimate assuming a 0.3g event is 8.07E-8/yr.

In addition to the prolonged loss of offsite power case, at higher accelerations the seismic forces result in component and/or structural concerns. For most safety-related components, the structures are not limiting and the impact can be based on component-level fragility. Reference 22 utilized existing seismic fragility information to arrive at a generic estimate for component capacities. A review of this report indicates that major equipment exhibits at least 1.0g median capacity given standard assumptions related to anchorage and location.

To develop an estimate for multiple component and/or structural seismic failures for WF3 a median capacity of 1.0g is utilized. The corresponding recurrence frequency of a seismic event of this acceleration or greater is 1.21E-6/yr. This is again multiplied by the probability of failures

at that acceleration ( 0.5) to arrive at a value of 6.07E-7/yr. This represents the frequency of core damage due to seismically-induced component and/or structural failures.

This estimate is considered a bounding contribution for seismically induced failures, because the probability of a seismically induced component failure associated with a seismic event of this magnitude would dominate postulated random failure probability. A typical assumption of one-fails-all-fail typically assumed for seismic faults would also tend to defeat redundant components and again lead to the conclusion that for this seismicity range the seismic failures would provide a reasonable estimate for the contribution to core damage and LERF.

Summing the estimates for lower acceleration seismic events which would be dominated by prolonged station blackout with the contribution from higher acceleration seismic events involving seismically induced component failures yields an estimated CDF contribution of 6.87E-7/yr (8.07E-8/yr + 6.07E-7/yr) and is controlled by higher acceleration seismic initiating event.

The findings contained in NUREG-1742 [15] indicate that the fire CDF is primarily determined by plant transient type of events such as those from assessed plant transients. The judgment is made based on this observation that it is reasonable to assume that the ratio of intact to impaired containments will be similar for fire as for the internal events such that the total CDF and the breakdown by EPRI Class will be equivalent to that presented for the internal events.

Since both fire and seismic are considered in this sensitivity study, the CDF contribution for fire is taken from the WF3 Fire PRA [18]. The value used in this study is the non-compliant fire risk evaluation CDF of 1.62E-5/yr.

Per the guidance contained in the EPRI report [2] the figure-of-merit for the risk impact assessment of extended ILRT intervals is given as:

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

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

$CDF_{FIRE}$ = 1.62E-5/yr (eq. 31)

$CDF_{SEISMIC}$ = 8.07E-8/yr + 6.07E-7/yr = 6.87E-7/yr (eq. 32)

Class 3b Frequency = [($CDF_{FIRE}$) + ($CDF_{SEISMIC}$)] * Class 3b Leakage Probability (eq. 33)

Class 3b Frequency = [(1.62E-5/yr) + (6.87E-7/yr)] * 2.3E-03 = 3.88E-8/yr (eq. 34)

No adjustment is made to the CDF values since LERF sequences are typically associated with SGTR or interfacing system LOCA sequences which are not represented by the external event assessments. This is potentially conservative, but is reasonable based on the simplified assessment, the conservative nature of the external events studies and the fact that many of the external event scenarios are long term station blackout and long term level of analysis detail. The change in LERF is estimated for the one (1) in ten (10) year and one (1) in fifteen (15) year cases and the change defined for the external events in Table 22.

Table 22
WF3 Upper Bound External Event Impact on ILRT LERF Calculation

| Hazard | EPRI Accident Class 3b Frequency | | | LERF Increase (from 1 per 10 years) |
|---|---|---|---|---|
| | 3 per 10 year | 1 per 10 year | 1 per 15 year | |
| External Events | 3.88E-8 | 1.29E-7 | 1.94E-7 | 6.47E-8 |
| Internal Events | 8.40E-9 | 2.80E-8 | 4.20E-8 | 1.40E-8 |
| Combined | 4.72E-8 | 1.57E-7 | 2.36E-7 | 7.87E-8 |

The internal event results are also provided to allow a composite value to be defined. When both the internal and external event contributions are combined the total change in LERF does not exceed the guidance for very small change in risk and does not exceed the 1.0E-7/yr change in LERF. The LERF increase supports the conclusion that the increased duration between tests does not result in a significant change in risk and the increase is acceptable per the criterion defined in the EPRI guidance document [2].

## 6.0 REFERENCES

1. Appendix J to Part 50 – Primary Reactor Containment Leakage Testing for Water-Cooling Power Reactors, U.S. Nuclear Regulatory Commission (USNRC), 10 CFR Part 50, Appendix J, January 2006.

2. Gisclon, J. M., et al, Risk Impact Assessment of Extended Integrated Leak Rate Testing Intervals: Revision 2-A of 1009325, Electric Power Research Institute, 1018234, October 2008.

3. Containment Integrated Leak Rate Test, Rev. 4 Change 9, Entergy Operations Incorporated, PE-005-001, August 2006.

4. Industry Guideline for Implementing Performance-Based Option of 10 CFR Part 50, Appendix J, Revision 3-A, Nuclear Energy Institute, NEI 94-01, July 2012.

5. WF3 Large Early Release Frequency (LERF) Model, Rev. 1, Entergy Operations Incorporated, PRA-W3-01-001S12, June 2009.

6. An Approach for Using Probabilistic Risk Assessment in Risk-Informed decisions on Plant-Specific Changes to the Licensing Basis, USNRC, Regulatory Guide 1.174, July 1998.

7. Reactor Risk Reference Document, Appendices J-O, Draft for Comment, USNRC, NUREG-1150, January 1987.

8. Park, C. K., et al, Evaluation of Severe Accident Risks: Zion, Unit 1, Rev. 1, USNRC, NUREG/CR-4551, Vol. 7, March 1993.

9. Waterford 3 Emergency Plan: Revision 45, Entergy Operators Incorporated, March 2014.

10. Sicard, P., Loss of Coolant Accident (LOCA) Alternative Source Term (AST) Radiological Dose Consequences for 3716 MWt Extended Power Uprate (EPU), Entergy Operations Incorporated, ECS04-001, August 2004.

11. Summitt, R., Comanche Peak Steam Electric Station Probabilistic Safety Assessment Evaluation of Risk Significance of ILRT Extension, Reliability and Safety Consulting Engineers Inc., RSC 01-47/R&R-PN-110, November 2001.

12. Dunn, D. S., et al, Containment Liner Corrosion Operating Experience Summary Technical Letter Report – Revision 1, USNRC, August 2011.

13. Sepelak, B., Containment Liner Through Wall Defect Discovered During Planned Visual Inspection, FirstEnergy Nuclear Operating Company (FENOC), LER 2013-002-01, February 2014.

14. Underwood, D., IPEEE Reduced Scope Seismic Margins Assessment (SMA) Waterford 3, Entergy Operations Incorporated, WF3-CS-12-00001, February 2012.

15. Perspectives Gained from the IPEEE Program, USNRC, NUREG-1742, April 2002.

16. WF3 at Power Fault Tree Model

| WF3 at Power Fault Tree Model | Size | Date |
|---|---|---|
| WF3Rev5.caf | 407 KB | 2/21/2013 11:07 am |
| WF3Rev5.rr | 6296 KB | 7/15/2014 6:32 pm |
| F-MASTR5.caf | 3 KB | 1/31/2013 10:07 am |
| recovery_rules4.txt | 27 KB | 1/31/2013 10:08 am |
| Mutex5 .txt | 28 KB | 1/31/2013 6:31 pm |
| COREDAMAGE.CUT | 7294 KB | 7/17/2014 1:48 pm |

17. Entergy Seismic Hazard Curve

| Entergy Seismic Hazard Curve | Size | Date |
|---|---|---|
| Entergy USGShazard.xlsx | 29 KB | 4/7/2011 1:18 pm |

18. Stephens, P., Comparison of Waterford 3 MOR and FRE CDF and LERF Results, Reliability and Safety Consulting Engineers, Inc., RSC-CALKNX-2013-0810, October 2013.

19. Letter to NRC from Calvert Clifffs Nuclear Power Plant Unit No.1. Docket No. 50-317, Response to Request for Additional Information Concerning the License Amendment Request for a One-Time Integrated Leakage Rate Test Extension, dated March 27, 2002.

20. RSC ILRT Excel Calculations for Waterford 3

| RSC ILRT Excel Calculations for Waterford 3 | Size | Date |
|---|---|---|
| ILRT Calculation Sheet WF3 (Deliverable).xlsx | 104.5 KB | 7/21/2014 |

21. Calculation of Reactor Accident Consequences: Appendix VI to Reactor Safety Study, USNRC, WASH-1400 (NUREG 75/014), October 1975.

22. Harrison, D., Generic Component Fragilities for the GE Advanced BWR Seismic Analysis, International Technology Corporation, International Technology Corporation, September 1988.

## Review Comments and Resolution

Reviewer Directions:

Provide detailed technical or global editorial comments here. Individual editorial or illustrative comments may be electronically provided (tracking) or attached to this review sheet.

Resolution Process:

Originator must provide resolutions for all comments.

Reviewer is to approve all proposed resolutions prior to completing the review process. No review is complete until this step is accomplished.

| Reviewer Comment | Originator Resolution of Comment | Reviewer Concurrence |
|---|---|---|
| Editorial comments provided in markup. | Updated report with all editorial changes. | RS |
| Page 3, the last bulleted item discusses “small” changes to CCFP but does not give any reference or actual baseline for comparison of what “small” is. Does such a metric exist?<br>In addition, suggest adding a discussion after Table 16 related to the CCFP results similar to what exists for the delta LERF metrics. | Added more information to the last bullet and after Table 16 detailing what the EPRI guidance document classifies as a small change in CCFP. | RS |
| It would be beneficial to a casual reader if some items were defined early in the report. Suggest defining what Type A, B, C testing are, as well as what EPRI Class 1, 2, 3, etc are. | Added a paragraph at the beginning of Section 1.1 that outlines the different type of containment leakage testing. The EPRI classifications are defined in Table 10 of the report. | RS |
| Table 6 header needs a reference filled in place of “XXX”. | Table 6 header title is now “Predicted Dose Rates from Reference 10” | RS |
| The short paragraph after Table 8 needs further explanation of how the calculation was performed as it is not possible to recreate it currently. | Added in an equation and calculation to clear up how the INTACT dose was developed. | RS |
| Should other noted assumptions throughout the report such as population evacuation levels be included in Section 3.0? | These are the assumption that the EPRI guidance document sets for the user. | RS |
| Consider moving some noted text from Section 4.1 into Section 1.1 to give more of the methodology up front. | The current formatting is approved by the NRC and will remain unchanged. | RS |

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|---|---|---|
| Equation #3 is not reproduced correctly in the supporting spreadsheet, updates are required which will slightly change the report's results. | Update was made to the excel spreadsheet and the report and all subsequent calculation and numbers are updated. | RS |
| Footnotes #2 and 3 in Table 12 are missing in the table text. | Added the superscripts in the correct locations of the table. | RS |
| Equations #13 and 14 require updates for the Class 3 probabilities which should match those presented in Equations #2 and 4 respectively. Similarly, Equations #19 and 20 require updates. | All mentioned equations have been updated | RS |
| The Jefferys Prior column in Table 20 requires updates for the Class 3 probabilities which should match those presented in Equations #2 and 4. Any changes to the results presented in Tables 20 and 21 from this update should also be made. | Tables 20 and 21 have been updated to reflect the correct Class 3 probabilities. | RS |
| What is the source for the 0.5 probability of loss of offsite power given a seismic event as discussed in Section 5.3? | The median capacity of LOSP is assumed to be 0.3g. Since this is a median capacity failure only occurs 50 percent of the time. Therefore a 0.5 multiplier is applied to the probability. | RS |
| Reference #18 is a duplicate of #1 and should be removed. | Removed reference | RS |
| The dose constant equation (Eq. 1) appears to be the inverse of that in the documentation in Reference 11. | Double checked the equation by hand against the reference the scaling factor is correct. | RS |
| The values for X and Y should be dose not dose rate since they are for an accumulation of so many hours. See Reference 11 Appendix C. | Change the values to be dose instead of dose rate. | RS |
| For Equation 19 and the like, you need to figure out some way to show that his is not zero because it is my by looking! Need a footnote or additional precision, something. | Change formatting from scientific to general number formatting with four significant digits. | RS |
| Added suggested text to highlight that WF3 is a low seismicity area. | Agreed and accepted the suggested text. | RS |

Entergy

Technical Review Comments and Resolutions Form

| | | | |
|---|---|---|---|
| Engineering Report Number | RSC 14-02 | Rev. Draft | Title Waterford 3 Evaluation of Risk Significance of an ILRT Extension |
| **Quality Related:** ☐ Yes ☒ No | | **Special Notes or Instructions:** | |

| Comment Number | Section/ Page No. | Review Comment | Response/Resolution | Preparer's Accept Initials |
|---|---|---|---|---|
| 1 | 2.0/4 | The Rev. 4 LERF model is used as the input. Since this rev is officially obsolete (superseded by Rev. 5 model), there should be a short discussion as to why the Rev. 4 model was used, e.g., Rev. 5 model results do not have the necessary resolution [LERF sequence results], Rev. 4 LERF is bounding [much higher] wrt to Rev. 5, etc. | The Rev. 5 model update scope was limited to LERF and did not update other containment outcomes needed for the ILRT evaluation. Therefore, the Rev. 4 model was deemed to be the most recent documentation that contained all of the necessary inputs to complete the ILRT analysis. | HAB |
| 1a | 2.0/4 | Response is good; please add this discussion to Section 2.0, as the comment requested. | Added this statement to the first paragraph of Section 2.0. Since the WF3 PRA Level 2 model was developed, updates have been performed to the internal events and LERF models, but the update did not perform calculations for the other containment endstates required for the ILRT extension evaluation. Therefore, the Rev. 4 model was deemed to be the most recent documentation that contained all of the necessary inputs to complete the ILRT analysis. | HAB |
| 2 | 2.0/4 | Last paragraph: WF3 model is given as reference [2], but [2] is the EPRI report. This should be [5]. Also, "[2]analysis" should be "[5] analysis", i.e., space after bracket. | Comment incorporated – fixed reference | HAB |
| 3 | Table 3/p.5 | Shouldn't frequency for LERF10 be "ε", as for the other sequences for which freq is 0? | Comment incorporated – fixed the one 0.00 frequency | HAB |
| 4 | 2.0/9 | In paragraph following Table 5, there is a statement "The | The dose values in Table 5 for EPRI Class 2, 7, and | HAB |

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| | | population dose factor is used to adjust the Zion population dose…" I can't find where this factor is used. Where is it used? | 8 are multiplied by the population dose factor to get the WF3 doses for these classes, which are listed in Table 9. | |
| 4a | 2.0/9 | I understand, but please make this clear in the document. For example, on p. 9, in the paragraph following "Population Dose Factor = 0.45", you could say that the more detailed dose calculation for INTACT is done because it provides the basis for the 3a and 3b doses that are key in the ILRT delta-dose. (Otherwise, what is the point?) In the last paragraph of p. 11, please expand upon "previously mentioned methods" to reiterate that the doses for Classes 2, 6, 7, and 8 use the population dose factor applied to the Zion doses. | Added additional text to the last sentence of the paragraph after population dose factor calculation.<br>Previous: While Zion had two release categories that fell into EPRI Class 1, a more accurate estimate for the INTACT dose rate at WF3 is developed using plant-specific data from Reference 10.<br>Updated: While Zion had two release categories that fell into EPRI Class 1, a more accurate estimate for the INTACT dose rate at WF3 is developed using plant-specific data from Reference 10. The INTACT dose is the basis for Class 3a and Class 3b doses, which are key in the ILRT delta-dose calculations. Therefore, using plant-specific information to develop the dose associated with INTACT yields results more reflective of the WF3 site.<br><br>The statement before Table 9 now reads: Table 9 contains the release category dose information. Class 1 dose information is derived from a scaling factor based on plant specific data. Class 2, Class 7, and Class 8 are developed by multiplying the Zion dose for these classes, contained in Table 5, by the population dose factor. Class 6 applies a decontamination factor of 0.1 to the dose associated with Class 2 based on an assumption that 10 percent of the release would be scrubbed. | HAB |
| 4b | 3.0/12 | Related to Comment 4a: add clarification to Assumption 7 such as, "Therefore, application of a simple Waterford population dose factor to the Zion doses for these classes is considered sufficient/adequate/reasonable/…something." | Text was added to the end of Assumption 7. " Therefore, the application of a simple WF3 population dose factor to the Zion doses for these classes is considered sufficient." | HAB |
| 5 | 2.0/10 | Table 6 uses the term "Dose Rates" in the title, but these are doses, not dose rates. Delete "Rates" from the Table 6 | Comment incorporated - "Rates" deleted from Table 6 | HAB |

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| | | title. | | |
| 6 | 2.0/pp.10 -11 | Tables 6, 7, and 8 use the units “person-rem”. This term is applicable to population doses (individual dose x population). Since the doses in these tables are individual doses calculated from the design basis LOCA analysis, the units here are “rem” rather than “person-rem”. “Person-rem” is appropriate for later in the calculation when these individual doses are multiplied by the population. | Comment incorporated - The correct unit of “rem” has been applied to these tables. | HAB |
| 7 | 2.0/11 | Paragraph following Table 8: as described in the previous comment, “person-rem” should be “rem”. [Note: in the next paragraph, person-rem is appropriate, i.e., 1.41E+4 person-rem.] | Comment incorporated with text change. | HAB |
| 8 | 2.0/11 | 2nd paragraph following Table 8: “The estimated population is 4.44E+6…” This is the Zion population. Why is the Zion population used? Shouldn’t the Waterford 3 population be used? I’ve gone over this calculation a number of times and can’t figure out why the Zion is used. | The Zion population should not have been used. Using the Waterford 3 population instead yields an INTACT dose estimate of 6.33E+3. | HAB |
| 9 | 2.0/11 | 2nd paragraph following Table 8: “However, it is usually assumed that ninety five (95) percent of the population will be evacuated…” Is there some source for this? Any sort of reference, such as a NUREG, EPRI report, canonical PRA? This is a bold statement to make without support, particularly since it reduces the population dose by a factor of 20. Since the evaluation of evacuation in a key component of a Level 3 PRA, there must be some references that will support this. | The source of this information is WASH-1400. A reference tag “[21]” has been added to this statement to support its validity. | HAB |
| 10 | Table 9/p.12 | How are the WF3 population dose values EPRI Classes 2, 6, 7, and 8 calculated? The Class 1 (INTACT) dose calculation is described in the preceding pages (but incorrectly using the Zion population?), but don’t understand how the others are calculated. Given that the 1.41E+4 is the intact dose calculated from the W3 LOCA dose calculation and extrapolated to 25 miles to represent an average dose for the 50 mile EPZ, I would expect that | Class 2, 7, and 8 are calculated based on the average exposure values listed in Table 5. These values are multiplied by the “population dose factor” to develop the WF3 dose listed in Table 9.<br>Class 2 = 5.97E+6 * 0.45 = 2.69E+6<br>Class 7 = 1.55E+8 * 0.45 = 6.98E+7<br>Class 8 = 1.26E+8 * 0.45 = 5.67E+7<br>Note that the values for Class 7 and 8 do not | HAB; addressing Comment 4 (and sub-comments) will take care of this comment. I |

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| | | the remaining WF3 dose valves would be calculated from the intact dose by ratioing from the Zion values, e.g., W3 Class 2 would be $Dose_{W3,Class\ 2} = Dose_{W3,Class\ 1} * (Dose_{Zion,Class\ 2} / Dose_{Zion,Class\ 1})$. It is not clear how the values in Table 9 are calculated, or even that they are correct; based on Zion doses in Table 5, I can't see how the Table 9 values could be correct. E.g., Zion Class 2 dose is factor of 8013 x the Zion Class 1 dose. W3 Class 2 dose is only a factor of 191 x the W3 Class 1 dose. How can this be? | directly match the values in Table 9, but are close. This is due to Excel carrying out all of the significant digits not contained in the simplified calculations above. Class 6 applies a decontamination factor to Class 2 of 0.1, which yields 2.69E+5.<br><br>The results are very dependent on Class 1; therefore, we used as much plant-specific data as possible to develop the INTACT dose. That is why the Zion surrogate is not used for INTACT. | understand what you did; what I'm really asking for with this comment is that the document make it clear what you did. I should have been clear about this: all of these comments that are questions like "how did you…?" are asking that the document make this clear. Entergy personnel need to be able to understand what was done, particularly people not originally involved. |
| 11 | 3.0/12 | Assumption 6: not sure what this means. Does it mean that the doses are integrations over an assumed post-accident time (e.g., 30 days in the LOCA analysis), and thus not time dependent on time? Please clarify. | The statement indicates that the release is constant and continuous and does not change with time. It is not related to duration since the duration is defined by the LERF definition provided in the PRA. | HAB |
| 11a | 3.0/12 | Please add this clarification to the assumption. | Assumption 6 now states: 6. The containment | HAB |

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| | | | releases are constant and continuous and are not impacted with time. The duration of the release is defined by the LERF definition provided in the PRA. | |
| 12 | 4.1/13 | Third bullet: Wrt "population dose rate", I know EPRI uses this terminology, but this is actually not a dose rate, it is a risk; that is population dose times annual frequency (LERF) = risk. Could this say "population risk (person-rem/yr)" instead of "population dose rate (person-rem/yr)"? | Since the EPRI guidance document (template) uses this terminology, the text is not changed to stay consistent with the accepted template. This is for ease of review when comparing the accepted template to the analysis. | HAB |
| 13 | 4.1/14 | Third paragraph says "The PRA did address Type A (liner-related) faults…"; could some text be added to make clear that this is talking about the Intact sequences in the LERF model, which are EPRI Class 1. I think that's what it means, correct? It could be clearer. | Yes this is talking about the INTACT sequences in the Level 2 model. Since a Type A leak falls under INTACT, 3a and 3b are derived as portions of the INTACT frequency in the calculations for the analysis. Added additional text to the sentence that references back to INTACT accident sequences. | HAB |
| 14 | Table 12/p.20 | Suggest combining the separated "Person-rem (calculated)" and "Person-rem (from La factors)" columns into a single column (such as in Table 11), since the separated columns don't add anything here. The separated columns would be useful in Table 11 (see next column). | Comment incorporated – Table 12 consolidated the Person-rem columns. | HAB |
| 15 | Table 11/p.16 | Suggest using the separated dose columns in Table 12 in Table 11 instead. This makes more sense because Table 11 is closer to being an input type table, while Table 12 is a calculation type table, which is repeated in Tables 13 and 14 for different test intervals. | In Table 11, the frequency for 3a and 3b has yet to be determined. It would not flow if the dose for these is listed and the frequency is not. Therefore, no change to Table 11 is made. | HAB |
| 16 | Table 12/p.20 | Footnotes to Table 12 have the wrong numbers. | Comment incorporated. | HAB |
| 17 | Tables 13, and 14/pp.22, 14 | Shouldn't footnote 2 refer to Table 12? Table 7 (in the footnote) is just the calculation parameters for C. | Yes, it should be referencing Table 12. Comment incorporated. | HAB |
| 18 | 5.2/33 | First paragraph is confusing. It's not clear that the expert elicitation was only done in the EPRI report. The EPRI sensitivity analysis and the application of that sensitivity analysis to the present calculation is described together in | The expert elicitation was only done in the EPRI report. Using the expert elicitation values for leakage occurrence instead of the Jefferys non-informative prior (baseline), the risk associated with | HAB |

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| | | this paragraph to the effect that it is not clear what is done in the present calculation vs. what is in the EPRI report. E.g., 1st sentence is "A second sensitivity case … is performed as described in the EPRI guidance document" followed by "In this sensitivity case, an expert elicitation was conducted…" This makes it seem that the present sensitivity case included an expert elicitation. Please revise this paragraph (and the second one) to make clear what was done. (If there was an expert elicitation performed in this calculation, it is not documented, and should be.) | extending the ILRT is calculated. Thus, the sensitivity is using the expert elicitation data from the EPRI guidance document as input into the calculation.<br><br>The first paragraph was reworded as follows:<br>"A second sensitivity case on the impacts of assumptions regarding pre-existing containment defect or flaw probabilities of occurrence and magnitude, or size of the flaw, is performed as described in the EPRI guidance document [2]. The expert elicitation contained in the EPRI report developed probabilities for pre-existing containment defects that would be detected by the ILRT only based on the historical testing data."<br><br>The second paragraph received no updates. The second paragraph is just summarizing what the experts did for EPRI and mentions that more details are found within the EPRI report. | |
| 19 | 5.3/35 | 3rd full paragraph: wrt to "a typical estimation for the median capacity of the offsite power supply (0.3g, median capacity)." Where is this from? I haven't been able to find this in the standard references, e.g., NUREG-1742, EPRI SMA report. | Added a reference tag "[22]" to the end of this statement. The reference is "Generic Component Fragility for the GE Advanced BWR Seismic Analysis". | HAB |
| 19a | 5.3/35 | If this really is a typical offsite power fragility, there must be some other, more accessible, reference for this besides a proprietary GE document. Isn't there a NUREG or EPRI report for this? We (Entergy-Waterford 3) really need a reference that we have access to. | This report is not proprietary. It was developed for the DOE. A copy of the reference will be provided with final document transmittal. Additionally NUREG/CR-4334 contains similar information for Zion fragilities with the EDG having a value of 1.06g and offsite power ceramic insulators having a value of 0.25g. The generic report was chosen as a reference over this document because it is a composite number based off more than one plant where as the values in the NUREG are Zion specific. | HAB |

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| 20 | 5.3/35 | 4th full paragraph: the 0.3g seismic event for W3 is given a probability of 3.60E-4/yr from Ref. 17 (Entergy_USGShazard.xlsx). How it this value derived? The Ref. 17 spreadsheet has the following PGA data for Waterford 3: 0.2272g => 1.79E-5 per yr; 0.3176g => 1.05E-5. How can the 0.3g prob. Be 3.6E-4/yr? | The incorrect base values were used for the interpolation calculation. Instead of the corresponding frequencies for 0.2272g and 0.3176g, the frequencies for 0.0215g and 0.0301g were used solving for 0.03g. Using the correct inputs, a value of 1.20E-5 is obtained. | HAB |
| 21 | 5.3/35 | Last paragraph: how is the 1.0g probability calculated? | The 1.0g is a representation of the median plant level fragility. This is comprised of components and structures. For plants similarly designed to WF3, structural failure has been shown to have a median capacity in excess of 2.0g and is risk insignificant. For components, the weakest links are typically associated with onsite power sources. At WF3, this is the EDGs. Generic estimates of EDG fragility provide a value of 1.0g as the median capacity. Therefore, the median capacity of the EDG is selected as the representative of the plant level fragility. Thus, when a seismic event of this magnitude occurs, there is a 50 percent chance the plant will suffer catastrophic damages. The frequency of occurrence (1.21E-6/yr) is multiplied by the CCDP (1.0) and 0.5 (since this is a median capacity). | HAB |
| 21a | 5.3/35 | Wrt to [22], is there a more accessible reference? See Comment 19a. | See response to 19a. | HAB |
| 22 | 6.0/38 | Reference 5 has the wrong calculation number: it should be PRA-***W3***-01-001S12. | Comment incorporated. | HAB |
| 23 | 2.0/11 | Wrt to the calculation of Y, following Table 8, when I use C=1.46, from Table 8, I get a Y value of 6.61E-2 vs 6.33E-2. When I calculate C from the Table 7 values, I get (to 5 significant figures): 1.4633; this then gives a Y value of 6.56E-2. Neither of these is close to the 6.33E-2 that is shown on p.11. | Edited the $d_{LPZ}$ miles to more accurately show 2.05 miles instead of the current 2.1. This number was not consistent with the 3 significant digits displayed in other areas of the calculation. Using these numbers in the calculation yields a value of 6.38E-2. This is again due to significant digits. Reference 10 gives a value of 2.4639 rem for $LPZ_{30d}$. This combined with the significant digits for C calculated in excel yields a value of 6.33E-2. | HAB |

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| 24 | Table 5/p.9 | In trying to confirm the exposures (doses) from NUREG/CR-4551 Vol. 7, I was unable to determine how the source term groups were determined. Could some explanation be added? E.g., reference to NUREG tables? | Added in a reference to Table 3.4-4 which breaks down the composition of the source term groups. This highest contributing release type for each source group was credited to the corresponding EPRI Class. | HAB |
| 24a | Table 5/p.9 | Reference to Table 3.4-4 of the NUREG is not enough. It is very difficult to figure out what was done to go from Table 3.4-4 to the groupings in Table 5. Please add some explanation (as the comment requested) about how this was done. For example, it would be helpful to say which Table 3.4-4 bins correspond to which EPRI classes; some of the bins codes in the table are very cryptic. The explanation in the response (highest contributing release type for each source group was credited to the corresponding EPRI Class) needs to be added. And, there is an implicit assumption that it is ok to just use the highest contributing bin, when in some cases the next contributor is quite significant and is in a different EPRI class; e.g., for Group 172, 0.52 is Alpha (large early?) and 0.37 is bypass (SGTR and V). This should be acknowledged and justified (at least recognized, since this group 172 case is probably an isolated example). | The following additional text has been added for clarification.<br>Table 3.4-4 in NUREG/CR-4551 [8] provides some guidance with respect to the composition of the source term grouping. The highest contributing release type was credited to the corresponding EPRI class. While multiple release types are contained in Table 3.4-4, only eight of the categories contained the majority of the release. Zion labeled these categories as Is. Leak, SGTR, LS, LL, EL, Alpha, NoCF, and BMT. Class 1 consists of any source term groups that are dominated by no containment failures (NoCF). EPRI Class 2 is related to isolation faults; therefore, source term groups with Is. Leak as the main contributor are placed into this EPRI class. EPRI class 7 is related to early and late phenomena-induced failures. Zion categories LS, LL, EL, Alpha, and BMT are all associated with these types of failures. EPRI Class 8 pertains to containment bypass. The Zion category associated with bypass is SGTR.<br>For some source term groups, the contributing type of release is not completely dominated by one single category but a mixture of categories all representing the EPRI classes. Occasionally, other contributors (excluding the highest contributor) make up a sizeable portion of the composition. These other contributors occasionally are types of releases that would be classified differently than the highest release contributor. An example is source term group 172, where the highest contributor is Alpha (Class 7), with 52 percent of the release, while the second and third highest are associated with bypass failures (Class 8), combining for 37 percent of the release.. This group was ultimately classified as | HAB |

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| | | | Class 7 because the Alpha release is considered the more severe type of release and was the highest contributor to the source term group. | |
| 25 | Table 9/p.12 | MINOR: Frequency for LERF+LATE should be 3.34E-6 (roundoff? I understand the spreadsheet carries lots of significant figures, but the result is that these table values end up being slightly off). | As in prior ILRT submittals the values compared are small and to represent the differences in the text has caused complexity and a lack of readability. We have found it best to provide sufficient values to support the overall results and have addressed the calculational accuracy by noting in Section 4.0 that "The detailed calculations performed to support this report were of a level of mathematical significance necessary to calculate the results recorded. However, the tables and illustrational calculation steps presented may present rounded values to support readability." To emphasize this the statement has been relocated to appear directly after the first paragraph in Section 1.0. Additionally a reference tag "[20]" was added to the first sentence to reference the excel file where the calculations are performed with significant digits. | HAB |
| 26 | Table 11/p.16 | MINOR: Frequency total should be 4.17E-6; Class 2 dose should be 4.41E-2; Class 7 dose should be 2.32E+2 (roundoff?) | The detailed assessment is correct. Please refer to the response to Comment #25. | HAB |
| 27 | 4.2/17 | Eq. 5 should be 0.5/218=0.0023 (i.e., 0.5/(217+1)=0.5/218) | Comment incorporated – Typo fixed | HAB |
| 28 | 4.2/17 | Eq. 4 result should be 3.37E-8 (roundoff?) | The detailed assessment is correct. Please refer to the response to Comment #25. | HAB |
| 29 | Table 12/p.20 | Person-rem values have roundoff errors, similar to those described in the previous comments. 4.03 vs. 4.02 (Class 1), 4.41 vs. 4.42 (Class 2), 4.76 vs. 4.75 (Class 3a), 4.73 vs. 4.74 (Class 6), 2.31 vs. 2.32 (Class 7). | The detailed assessment is correct. Please refer to the response to Comment #25. | HAB |
| 30 | 4.2/20 | In $Total_{BASE}$ and $\%Risk_{BASE}$ calculations, the total person-rem value used should be 260, not 260.0016. Although 260.0016 may be the value from the spreadsheet, it is not consistent with the 3 significant figures (2.60[E+2]) in Table 12, from which it is taken. The extra figures are not needed in the calculation and should be removed. | Comment incorporated - The numbers have been changed to2.60E+2. | HAB |

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| 31 | 4.2/22 | Table 13 also has some slight roundoff errors in some of the Person-rem values. Also, the same comment as the previous one regarding 260.0388. | The detailed assessment is correct. Please refer to the response to Comment #25. | HAB |
| 32 | 4.2/23 | Since the extra digits are needed for calculating the very small difference between the two total person-rem values, it is of course correct to include them, but it might help to note that these values are the sum of the individual values in the tables, and why this is ok. (In typical engineering/scientific calculations such a summation is mathematically invalid: in a summation, when the decimal points are lined up the significance of the sum is equal to the significance of the lest sum (in terms of decimal places), i.e., the uncertainty of the larger values overwhelms the smaller values. In the present calculation, only the 3a and 3b (and the adjusted 1) person-rem values are actually changing, so the delta values are really just a function of these 3a and 3b values. Using the total person-rem values is sort of confusing. | The guidance document asks for an evaluation of the risk impact (in terms of population dose rate and percentile change in population dose rate) for the interval extension cases. These calculations are performing the second requirement in that statement. If the percentile change only used INTACT (Class 1), Class 3a, and Class 3b as inputs into the equation it would not accurately represent the change in dose rate because it does not capture the entire population dose rate. An additional requirement in the guidance document is tracking the change in LERF (Class 3b) and CCFP. These two are addressed later in the report.<br><br>However, text changes were made to these calculations to better convey that the increase is due to these three classes driving the total population dose rate up. Previously the extra significant digits were used to provide sufficient accuracy to avoid erroneous conclusions based on a lack of change represented by the calculation. Now the text adheres to the 3 significant digits formatting in the report. The calculations for percentile change are no longer presented in terms of totals. Now the equations are presented as: $Delta\%Risk_{10} = [((Class1_{10} + Class3a_{10} \text{-}+ Class3b_{10}) – (Class1_{BASE} + Class3a_{BASE} + Class3b_{BASE}))/ Total_{BASE}] \times 100.0$. This allows the reader to better understand that all the change is happening because of extending Type A testing. | HAB; it would have been easier to just add some explanatory text, but I think the comment was not clear. Adding the additional calculational detail is ok. |
| 32a | 4.2/pp. 23, 25 | In the added text, the person-rem value for Class 3b base from Table 12 is 5.31E-3. | Fixed the typos associated with Class 3b base values. | HAB |
| 32b | Table 12 | For Class 7 Description, “induced” was changed to “Induce”; should be “Induced”. Or better yet, “…Phenomena-induced” | Comment incorporated – texted changed to “Phenomena-induced” | HAB |
| 33 | 4.2/pp.24-25 | Ditto on the Table 14 roundoff errors and 260…. digits. | The detailed assessment is correct. Please refer to the response to Comment #25. | HAB |

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| 34 | 4.2/25 | In the paragraph beginning “From the EPRI Report”, the allowable La value is by implication 1.31E+4. Where does this come from? I can’t see this an input anywhere. | To remove any confusion with Licensing Design Basis the word “allowable” has been removed. The existing wording was taken from the accepted EPRI template. | HAB |
| 35 | Table 1/p.2 | Where do the “Type A testing risk (person-rem/yr)” values in the 2nd row come from? Shouldn’t these come from Tables 12, 13, and 14? E.g., the base Type A risk in Table 1 is shown as 1.66E-2, while in Table 12 it is 1.18E-2 (Class 3b). | The values do come from Tables 12, 13, and 14. Type A testing is the combination of Class 3a and Class 3b. Combining Class 3a (4.75E-3) and Class 3b (1.18E-2) from Table 12 yields a Type A testing risk value of 1.66E-2. Class 3b is only the LERF contribution to Type A testing. | HAB |
| 35a | Table 1 | Could you please state in the text above Table 1 that the Type A testing risk is a combination of Classes 3a and 3b? It just makes things clearer. The statement “The Type A contribution to large early release frequency (LERF) is defined as the contribution from Class 3b” can cause the reader to think the Type A risk is also form Class 3b, without digging into the details of the calculation (and I missed it even after digging into the calculation—it’s complicated and difficult to keep everything straight until you become very familiar with it.) | Changed text in the paragraph above Table 1 to expand on Type A composition.<br>Previous: The Type A contribution to large early release frequency (LERF) is defined as the contribution from Class 3b.<br>Updated: Type A testing risk is comprised of EPRI Class 3a and Class 3b. Class 3b is defined as the large early release (LERF) contribution to Type A testing. | HAB |
| 36 | Table 1/p.2 | The delta-risk from current and from baseline (2.66E-2 and 6.38E-2) do not agree with the Type A risk values in the 2nd row of the table. | This is a delta between the Total integrated risk not the Type A testing risk. This would explain the numbers not matching up. | HAB |
| 37 | Table 16/p.27 | Again, slight roundoff differences. But the delta-CCFP for 15 yrs vs. baseline, 8.05E-3, does not agree with the delta of 9.32E-1 minus 9.23E-1. From these value, the delta should be 9E-3, a significant difference. (The 8.05E-3 is also in the text below the table.) | The detailed assessment is correct. Please refer to the response to Comment #25. | HAB |
| 38 | Table 1/p.2 | The delta-CCFP for 15 yrs vs. baseline is 8.05E-3, and should be 9E-3. | The detailed assessment is correct. Please refer to the response to Comment #25. | HAB |

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| Verified/Reviewed By: | Howard Brodt | | Date | 8-6-14 | Resolved By: | Stephen Pionke |
| Site/Department: | Waterford 3 PRA | Ph. | 601-331-4651 | | Date: 8-8-14 | |

**Attachment 7 to**

**W3F1-2014-0052**

**List of Regulatory Commitments**

List of Regulatory Commitments

This table identifies actions discussed in this letter for which Entergy commits to perform. Any other actions discussed in this submittal are described for the NRC's information and are not commitments.

| COMMITMENT | TYPE (Check one) | | SCHEDULED COMPLETION DATE (If Required) |
|---|---|---|---|
| | ONE-TIME ACTION | CONTINUING COMPLIANCE | |
| WF3 will use the definition in Section 5.0 of NEI 94-01, Revision 2-A, for calculating the Type A leakage rate when future WF3 Type A tests are performed. | | X | Following the NRC approval of this license amendment request. |
| The normal Type A test interval should be less than 15 years. If a licensee has to utilize the provisions of Section 9.1 of NEI TR 94-01, Revision 2, related to extending the ILRT interval beyond 15 years, the licensee must demonstrate to the NRC staff that it is an unforeseen emergent condition. | | X | Following the NRC approval of this license amendment request. |