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10 CFR 50.90

W3F1-2015-0021

May 4, 2015

U.S. Nuclear Regulatory Commission
Attn: Document Control Desk
11555 Rockville Pike
Rockville, MD 20852

Subject: Waterford Steam Electric Station, Unit 3 Response to Request for Additional Information Regarding the Request to Permanently Extend the Integrated Leak Rate Test Frequency to 15 Years
Waterford Steam Electric Station, Unit 3 (Waterford 3)
Docket No. 50-382
License No. NPF-38

REFERENCES:

1. Entergy Letter W3F1-2014-0052, License Amendment Request to Change Technical Specifications to Extend the Type A Test Frequency to 15 Years, dated August 28, 2014. (ADAMS Accession No. ML14241A305)
2. Letter from NRC, Request for Additional Information Regarding the Request to Permanently Extend the Integrated Leak Rate Test Frequency to 15 Years (TAC No. MF4727), dated February 18, 2015. (ADAMS Accession No. ML15033A422)

Dear Sir or Madam:

In letter dated August 28, 2014 (Reference 1), Entergy Operations, Inc. (Entergy) submitted a license amendment request to change the Waterford 3 Technical Specifications to permanently extend the Integrated Leak Rate Test (ILRT) frequency to 15 years.

In letter dated February 18, 2015 (Reference 2), NRC requested Entergy to provide additional information to support review of the license amendment request to extend the ILRT frequency. This letter provides the response to that request for additional information.

This correspondence contains no new commitments.

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

I declare under penalty of perjury that the foregoing is true and correct. Executed on May 4, 2015.

Sincerely,

for M. Chisum

MRC/LEM

Attachments: 1. Waterford 3 Response to Request for Additional Information (TAC No.MF4727)

2. Internal Events PRA Peer Review – Facts and Observations (Findings Only)

3. Calculation, Waterford 3 Evaluation of Risk Significance of an ILRT Extension

4. Revised Section 4.5.3 of License Amendment Request

cc: Mr. Marc L. Dapas, Regional Administrator
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Attachment 1 to

W3F1-2015-0021

Waterford 3 Response to Request for Additional Information dated February 18, 2015.

(TAC No.MF4727)

By letter dated August 28, 2014 (Agencywide Documents Access and Management System (ADAMS) Accession No. ML14241A305), Entergy Operations, Inc., submitted a license amendment request (LAR) to change the Waterford Steam Electric Station, Unit 3 (WF3) Technical Specification 6.15, "Containment Leakage Rate Testing Program," to allow a permanent extension of the Type A primary containment integrated leak rate test frequency from 10 years to 15 years.

The U.S. Nuclear Regulatory Commission staff has reviewed the LAR and the following information is needed to complete the review**.**

RAI #1

Regulatory Issue Summary 2007-06 states that the NRC staff expects that licensees fully address all scope elements with Revision 2 of Regulatory Guide (RG) 1.200, "An Approach for Determining the Technical Adequacy of Probabilistic Risk Assessment Results for Risk-Informed Activities," by the end of its implementation period (i.e., one year after the issuance of Revision 2 of RG 1.200). Revision 2 of RG 1.200 endorses, with exceptions and clarifications, the combined American Society of Mechanical Engineers (ASME)/American Nuclear Society (ANS) PRA standard ASME/ANS RA-Sa-2009, "Standard for Level 1/Large Early Release Frequency Probabilistic Risk Assessment for Nuclear Power Plant Applications."

Given that the implementation date of RG 1.200, Revision 2, was April 2010, and the LAR was submitted in September 2014, identify any gaps between the WF3 internal events PRA model used in this application and RG 1.200, Revision 2, requirements that are relevant to this LAR. Additionally, address the technical adequacy requirements of RG 1.200, Revision 2, that are applicable to this LAR, or explain why addressing the requirements would have no impact on this application.

RAI #1 Response

The internal events PRA model used in the baseline analysis was the Revision 4 Internal Events PRA model which is the model that underwent a RG 1.200 Rev. 1 Peer Review. The Revision 5 model was not ready for use in this application because the Level 2 portion (damage states other than LERF) was not complete at the time of the LAR submittal. Since then, the Level 2 portion of the Rev. 5 model has been completed and a sensitivity analysis was performed to address the impact of using the updated analysis . The results of this sensitivity show that, although some risk increase occurs with the update, all risk metrics still meet the acceptance criteria for acceptable risk thresholds. Since this is the case, the technical adequacy of the internal events PRA as it is applicable to this application is based on the Revision 5 model.

The Waterford 3 PRA (Revision 4) has undergone a RG 1.200 Rev. 1 Peer Review against the ASME PRA Supporting Requirements by a team of knowledgeable industry (vendor and utility) personnel. The review was conducted by the Westinghouse Owners Group in August of 2009. The conclusion of the review was that the Waterford 3 PRA model substantially meets the ASME PRA Standard and can be used to support risk-informed applications.

The findings and conclusions of this review are contained in LTR-RAM-II-09-039, "RG 1.200 PRA Peer Review Against the ASME PRA Standard Requirements for the Waterford Steam Electric Station, Unit 3 Probabilistic Risk Assessment." The overall conclusion found that the

Waterford 3 PRA meets the ASME PRA Standard at Capability Category II or better for 81% of the applicable Supporting Requirements, with 90% met at Capability Category I or better. This review resulted in ninety-six new Facts and Observations (F&Os), forty-nine “Suggestions”, forty-five “Findings” and two “Best Practices”. Overall, the Waterford 3 PRA was found to substantially meet the ASME PRA Standard at Capability Category II and can be used to support risk-informed applications.

Since the completion of this Peer Review, Reg. Guide 1.200 was revised to Revision 2 which endorses the ASME/ANS PRA Standard RA-Sa-2009. Because of this revision, a Gap Assessment was performed to determine if the results of the Peer Review would have been altered if the later issue of the Reg. Guide were to have been used (PSA-WF3-08-01). The result of this Gap Assessment shows that no additional Findings would have been issued however two Suggestion level F&Os could potentially have been considered as Findings.

Moreover, since the Peer Review was performed in 2009, the Waterford Internal Events PRA model has been updated (to Revision 5) in support of efforts to transition to a risk-informed licensing basis under NFPA-805. While no changes in methods were associated with this update, most of the open Findings were addressed. Although this model was not used in performance of the original RI-ILRT, a sensitivity study has been performed to see how the results presented in this License Amendment Request are sensitive to the updated model (ECS14-010, Rev. 1, “Waterford 3 Evaluation of Risk Significance of an ILRT Extension”, contained in Attachment 3 of this letter). As can be seen in Table 1 below, usage of the updated model causes a slight increase in the resulting risk metrics, but the change in LERF is still within the Reg. Guide 1.174 guidelines for a “very small” change and the percent change in CCFP is still below the 1.5% criterion.

Table 1: ILRT Extension Risk Changes—Baseline and Revision 5 Model

Changes due to extension from 10 years (current)		Baseline	W/ Rev. 5
Δ Risk from current (Person-rem/yr)		2.01E-02	2.74E-02
% Increase from current (Δ Risk / Total Risk)		0.006%	0.007%
Δ LERF from current (per year)		2.35E-08	3.20E-08
Δ CCFP from current		3.53E-03	3.79E-03
Δ CCFP from current (% Change)		0.46%	0.57%
Changes due to extension from 3 years (baseline)			
Δ Risk from baseline (Person-rem/yr)		4.82E-02	6.57E-02
% Increase from baseline (Δ Risk / Total Risk)		0.014%	0.017%
Δ LERF from baseline (per year)		5.64E-08	7.68E-08
Δ CCFP from baseline		8.47E-03	9.08E-03
Δ CCFP from baseline (% Change)		1.10%	1.39%

The variations displayed in the above table show the impact that the resolution of the Findings had on the risk calculation performed for this LAR. As is evident from the table the population dose, delta LERF, and percentage CCFP change metrics still meet the acceptance criteria described in the LAR. However, since not every Finding was completely addressed in the model update, potential impact to this LAR could exist. Therefore, all Findings are discussed in Attachment 2 to this letter to show their related Supporting Requirement and disposition related to this LAR. The two Suggestion level F&Os identified in the Gap Analysis are also included in Attachment 2.

It should be noted that the Internal Floods Hazard Group was not included in the original assessment. In order to address the findings related to IF, it is necessary to gain insights into the impact this hazard group has on the results. Waterford has an Internal Floods model that was peer reviewed along with the Internal Events model and has a total CDF of 2.48E-6; however, it does not assess all Level 2 end states. The risk calculation supporting the LAR was revised to include this contribution and is attached as Attachment 3 to this letter. Because these changes affected statements made in the LAR, Attachment 4 to this letter includes the applicable changes to the LAR resulting from this calculation revision. (Since no conclusions made in the LAR were changed by including the Internal Floods contribution, only Section 4.5.3 required revision.)

The results provided in Table 1 above already include the contribution from Internal Floods for both the baseline and the updated model case. With the inclusion of the Internal Flood contribution, the Reg. Guide 1.200 guidance is met for inclusion of necessary hazard groups. All applicable hazards groups have been addressed by the analysis including Internal Events, Internal Floods, Internal Fires, and Seismic.

As described by the dispositions to the F&Os in Attachment 2, most of the peer review findings have already been addressed in the Rev. 5 model. For those that were not addressed, the impact to this LAR would not be significant had the items been addressed in the model. An additional sensitivity case was performed to determine the impact to the results by doubling the Internal Floods contribution. The results of this sensitivity case showed that the risk thresholds for population dose, delta LERF, and percentage CCFP change were not exceeded. This provides further confidence that addressing open findings would not cause an adverse impact on this application. Resolution of Peer Review gaps is discussed in more detail in the response to RAI #3.

The primary piece of the PRA used to support this application is the Internal Events Model along with the contribution from Internal Floods. At-Power operation is the only operational mode needed for consideration since containment is opened for the majority of shutdown operations and therefore, leakage would be of minimal concern. Also, the resolution of most of the Peer Review Findings and the respective minimal impact as described in Attachment 2 demonstrates the technical adequacy of the PRA used in this application.

The key assumptions and key sources of model uncertainty as they relate to this application originate mainly from the Level 2 analysis and from the EPRI methodology used to determine the risk increase associated with the test frequency extension. A couple specific items should be mentioned:

1. One of the key assumptions from the EPRI methodology is that all Class 3b releases would be categorized as LERF based on the NEI guidance. Since LERF is used directly as the risk metric, and because not all leakage related Class 3b releases would be LERF, this is considered as a conservative assumption (ECS14-010, R1).
2. Because the Internal Floods analysis did not calculate Level 2 damage states after core damage, it was assumed that half the CDF contribution would bin to Intact and half would bin to LERF. This is consistent with the contribution from the sequence results in the revised Level 2 analysis (PSA-WF3-01-LE Rev. 1, “WF3 Large Early Release Frequency Model”) for transients which is the only initiator used in the Internal Floods analysis.
3. Another key assumption and source of model uncertainty is related to the simplified Level 2 analysis used in the Revision 4 PRA model which follows the Westinghouse guidance from WCAP 16341 P, “Simplified Level 2 Modeling Guidelines.” This guidance results in a larger contribution to LERF because of simplifications. The updated Internal Events model (Revision 5) did a more detailed analysis to address some of these simplifications and the result was a lower LERF contribution, but a higher INTACT which leads to higher Class 3b leakage LERF contribution. The impact of this assumption is seen in the sensitivity analysis using the Revision 5 model and the slight increase in risk results because of this higher INTACT contribution.

Based on the discussion above, it can be concluded that the PRA model used to support this application is of sufficient quality per the guidance contained in Regulatory Guide 1.200 Revision 2. The sensitivity case using the Revision 5 model shows the impact of addressing the majority of the peer review findings while the sensitivity case to increase the Internal Flooding contribution shows that any findings that remain open would not impact the conclusions included in the original LAR.

RAI #2

Section 4.5.2 of the LAR states that, “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.” The ASME PRA Standard RA-Sa- 2009 contains 10 parts, each with several sections. Clarify whether the above statement from the LAR refers to Sections 2 and 3 of Part 4, “Requirements for Fire At-Power PRA.” If the Fire PRA has not been peer-reviewed against ASME/ANS RA-Sa-2009, clarify how the fire PRA was determined to be of sufficient quality for this application.

RAI #2 Response

The correct verbiage in the LAR for statement in question above should have been to refer to Section 4 of the 2009 ASME PRA Standard. The PRA quality of the Internal Events PRA model (which is used as an input into the Fire PRA model) is shown by the Peer Review against Sections 2 and 3 of the ASME PRA Standard. However, the technical elements in Section 4 of the ASME Standard cover the full breadth of the Fire PRA. The Waterford Fire PRA has been peer-reviewed against Section 4 (Part 4) of ASME/ANS RA-Sa-2009. Specifically, the Fire PRA peer review used the Supporting

Requirements (SRs) in Section 4 of ASME/ANS PRA Standard along with any associated NRC clarifications or qualifications for the individual SRs as contained in Revision 2 to RG 1.200.

RAI #3

Section 4.5.2 of the LAR states that, "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." Provide a list of all supporting requirements from the peer-review relevant to this LAR for which the PRA did not meet the ASME/ANS RA-Sa-2009 capability category 1 supporting requirements. Explain why these gaps would not impact this specific application. For gaps that did not impact another application (e.g., NFPA-805) describe why the finding does not impact this LAR.

RAI #3 Response

As discussed in RAI Response #1, the original baseline analysis used in this application utilized the Waterford 3 Internal Events PRA model (Revision 4) that underwent the Peer Review, and not the updated model (Revision 5) which resolved most of the Findings from the Peer Review. The Fire PRA model used for the external events analysis did utilize the updated Internal Events PRA model (Revision 5). However, as also discussed in RAI #1, a sensitivity analysis was performed (ECS14-010 Rev. 1, "Waterford 3 Evaluation of Risk Significance of an ILRT") using the updated PRA model (Revision 5) and found that, although the risk results showed some increase, all risk criteria are still met. Since these criteria are met and the Rev. 5 Level 2 model is now the model of record (it was completed shortly after submittal of this LAR), the remaining gaps that are relevant to this LAR are only those related to the updated model.

The Internal Events PRA model underwent a peer review against the ASME/ANS PRA Standard RA-Sb-2005 as clarified by RG 1.200, Rev. 1. Based on the gap analysis done in PSA-WF3-08-01 ("Waterford 3 PRA Peer Review Gap Assessment to the 2009 PRA Standard"), no additional gaps were found between the Internal Events PRA model and the ASME/ANS PRA Standard RA-Sa-2009 as endorsed by RG 1.200, Rev 2. However, this report did note that two Facts and Observations (F&Os) that were originally given as "Suggestions" would probably be considered as "Findings" if using Revision 2 to RG 1.200.

The Internal Events Peer Review report (LTR-RAM-II-09-039, "RG 1.200 PRA Peer Review Against the ASME PRA Standard Requirements for the Waterford Steam Electric Station, Unit 3 Probabilistic Risk Assessment") lists the assessment of Supporting Requirement Capability Categories (CCs). Of all the SRs, 31 did not meet CC-I, while the remaining SRs were evaluated as meeting CC-I or greater. Each SR that did not meet CC-I has corresponding F&Os related to the finding. The update to the Internal Events model was done to incorporate plant changes and to address the F&Os given from the Peer Review. Attachment 2 lists the F&Os classified as Findings as well as the two Suggestion F&Os identified in the Gap Analysis. A disposition for each of these F&Os is also given to describe either how the finding was resolved or its applicability to this application. This table excludes the MU (Configuration Control) SRs as they have no impact on this application.

Based on the dispositions of these F&Os, only 9 Internal Events specific F&Os would still be considered as not fully addressed and 8 Internal Floods specific F&Os would still be considered as not addressed. These F&Os relate to SRs that did not meet CC-I. Based on the dispositions of the Internal Events F&Os in Attachment 2, only two Findings (AS-A7-01 and SC-B3-01)

would have potential non-negligible impact on this application if they were resolved. For the F&Os related to Internal Flooding, based on the dispositions in Attachment 2, resolution of two of these Findings (IF-B2-01 and IF-D7-01) would have potential non-negligible impact on this application.

Due to other conservatisms in the Internal Flooding model such as conservative treatment of flood mitigating operator actions and a bounding duration of flooding release, any potential impact because of these findings would be greatly reduced if not negated. Qualitatively speaking, more detailed operator recovery action credit itself would provide reduction in CCDP for many of the flooding scenarios.

For the two F&Os related specifically to Internal Events, only two have the potential to impact this LAR. However, though the potential impact would not be negligible, it would be bounded by the sensitivity analyses performed. The sensitivity case showing the impact of doubling the internal floods contribution shows that nearly a 40% additional increase in CDF with respect to the Internal Events CDF (the CDF in relation to the risk calculation is equivalent to the sum of Level 2 Plant Damage states) would not cause the risk thresholds associated with this application to be exceeded. The binning of this contribution was half INTACT and half LATE which is more conservative than the SBO (both internal events gaps are related to SBO sequences) contribution to Level 2 damage states of less than 25% INTACT.

Therefore, as discussed in the dispositions in Attachment 2, the bounding impact related to the SC-B3-01 Finding would be a 25% increase in CDF while the AS-A7-01 Finding would be much less than that. So, it can be concluded that the resolution of these Findings would not have any adverse impact on this application and that additional margin exists for any potential impact from Internal Flooding gaps. Also, it should be noted that reference to other applications is not included in Attachment 2, and each gap is discussed with respect to this LAR.

RAI #4

In the LAR, the licensee proposed to revise Section 6.15 of WF3 TS, as follows:

> 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 term “except that” in the above proposed TS wording gives the appearance that the extension of the next Type A test is an exception to the guidelines contained in NEI 94-01, Revision 2A. Provide clarification for the term “except that.”

The NRC staff notes that this was identified for similar applications previously submitted for the NRC review and Entergy had provided clarification in letters dated January 20, 2011, for Arkansas Nuclear One, Unit 2, and March 11, 2014, for Arkansas Nuclear One, Unit 1.

RAI #4 Response

Entergy is not requesting any exceptions to the guidelines contained in NEI 94-01, Revision 2-A. The term “except that” in the proposed TS wording of the revision are removed. The proposed revision to Section 6.15 of W3 TS is as follows:

> 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. The next Type A test performed after the May 21, 2005 Type A test shall be performed no later than May 20, 2020.

RAI #5

Sections 4.0 and 4.3 of the LAR state that the ASME Boiler and Pressure Vessel (BPV) Code, Section XI, Subsection IWL, does not apply to WF3.

As described in Section 3.8 of the WF3 final safety analysis report, both the shield building and the containment vessel are supported on a common reinforced concrete foundation mat. The containment vessel is supported on the concrete fill, which transfers the loads by bearing to the foundation mat below.

Subsection IWL provides the examination requirements for reinforced concrete Class CC components. Considering that the containment vessel is supported on a concrete fill and a reinforced concrete foundation mat, provide clarification regarding the LAR’s statement of Subsection IWL not being applicable to WF3.

RAI #5 Response

Subsection IWL provides the examination requirements for reinforced concrete Class CC components. Although the containment vessel is supported on a concrete fill and a reinforced concrete mat, it is not part of the containment system.
Per ASME Section XI 2001- 2003 Addenda, Subsection IWL-1210 Examination Requirements, “The examination requirements of this Subsection shall apply to concrete containments.”

Per WF3 FSAR, Section 3.8.1, Concrete Containment, “The Containment System does not utilize a concrete containment. The primary containment is a free standing steel pressure vessel which is surrounded by a reinforced concrete Shield Building. The Shield Building is designed as a seismic Category I structure and is discussed under Subsection 3.8.4. The steel containment and the Reactor Building internal structure are described in Subsection 3.8.2 and 3.8.3, respectively.”

"The Steel Containment Vessel (SCV) is a low leakage rate free standing steel pressure shell, completely enclosed by the concrete shield structure, with an annular space provided between the walls and domes of each structure to permit construction, operations, and in-service inspection. The SCV consists of a vertical upright cylinder, all welded steel pressure vessel, with hemispherical top head and an ASME ellipsoidal bottom head. The steel vessel is rigidly supported on a concrete base that was placed after the cylindrical shell and the ellipsoidal bottom had been constructed and post weld heat treated. The containment vessel, shield building, reactor auxiliary building, and fuel handling building are supported on a common foundation mat. Concrete floor fill was placed above the ellipsoidal shell bottom of the SCV after the vessel had been post weld heat treated, to anchor the vessel. All components and framing inside the SCV are supported on the concrete floor fill."

Per ASME Section XI 2001 – 2003 Addenda, Subsection IWL-1220(b), portions of the concrete surface that are covered by the liner, foundation material, or backfill, or are otherwise obstructed by adjacent structures, components, parts, or appurtenances, are exempt from the examination requirements of IWL-2000. Per ASME Section XI 2001 – 2003 Addenda, Subsection IWE-1220(b), embedded or inaccessible portions of containment vessels, parts, and appurtenances that met the requirements of the original Construction Code are exempted from the examination requirements of IWE-2000. Since the common concrete foundation slab and the bottom steel plate are inaccessible, they are exempt from examination per ASME Section XI 2001 – 2003 Addenda, Subsection IWL-1220(b) and IWE-1220(b) respectively.

RAI #6

Please provide information of instances, during implementation of the WF3 containment in-service inspection program, where existence of or potential for degraded conditions in inaccessible areas were identified and evaluated based on conditions found in accessible areas, as required by 10 CFR 50.55a(b)(2)(viii)(E) and 10 CFR 50.55a(b)(2)(ix)(A). If there were any instances of such conditions, discuss the findings and corrective actions taken to disposition the findings.

RAI #6 Response

A condition report dated 10/20/2000 documents an instance, during implementation of the WF3 containment in-service inspection program, where existence of or potential for degraded conditions in inaccessible areas were identified and evaluated based on conditions found in accessible areas. The condition description states that:

"VT-3 Examinations of the interior moisture barrier (located between the containment vessel and the concrete floor on the ledge at elevation - 1.5') revealed 22 locations where the moisture barrier has failed by various mechanisms. The moisture barrier is intended to provide long term corrosion protection to the containment vessel. No immediate/short term challenges to containment integrity were noted during the examinations. The NDE visual examination report provides detail on the location and conditions noted. Additionally, the affected areas have been marked on the containment vessel.

One of the affected locations is located immediately below penetration #21. This location is being wetted by condensation from the CCW pipe. The containment vessel at location #21 is experiencing general corrosion. The corrosion noted is not sufficient to affect either the structural integrity or the leak tightness of containment; however, the corrosion does indicate the potential for degradation below the moisture barrier and requires further investigation.

None of the remaining locations exhibited signs of either wetting or corrosion of the containment vessel."

The Responsible Engineer's (RE) Evaluation of Inaccessible Areas was documented in the response to the corrective action dated 1/30/2001 and is listed below:

Scope:

This evaluation covers the evaluations required by CEP-CII-002 paragraph 1.7.3.5 and by 10 CFR 50.55a(b)(2)(ix)(A). Evaluations that are required by CEP-CII-002 paragraph 1.7.3.3 are documented in attachment 2 to this corrective action (CA).

Results of Evaluation:

1) During examination of the moisture barrier two areas were identified which could indicate the presence of degradation in inaccessible areas.
2) Investigation of the first area revealed only limited areas of surface corrosion with no significant wall loss or pitting. All surface areas of the containment vessel at this location were determined to be acceptable by examination in accordance with IWE-3122.1.
3) Investigation of the second area revealed excessive corrosion in the region below the moisture barrier in the annulus. A condition report dated 10/27/2000 was prepared to document corrective actions associated with this corrosion.

Discussion:

Paragraph 1.7.3.5 of CEP-CII-002 requires the RE (or designee) to prepare a condition report when the RE determines that conditions exist in accessible areas which could indicate the presence of or result in degradation of inaccessible areas. The purpose of this evaluation is to evaluate the acceptability of the inaccessible area in question. Additionally the RE is to prepare inputs to the OAR-1 which include the following:

1) A description of the type and estimated extent of degradation, and the conditions that led to the degradation;
2) An evaluation of each area, and the results of the evaluation, and;
3) A description of necessary corrective actions.
   CA #7 addresses the need for the RE to provide inputs to the OAR-1

The flaws identified by the NDE VT-3 reports revealed two areas that indicated potential degradation of the containment vessel in the inaccessible areas below the moisture barrier.

One area is located immediately below penetration #21 and has been wetted due to condensation from the CCW pipe using penetration #21. General corrosion of the containment vessel was noted in the vicinity of the moisture barrier in this location. After removal of the moisture barrier, a small area of general corrosion was noted to exist below the moisture barrier at this location. This area of corrosion did not extend below the area that could be accessed by removal of the moisture barrier. At this location, the corrosion consisted of only a light surface corrosion with no pitting or cracking. Additionally, there was no discernable thinning of the containment vessel due to the corrosion. As a result the corrosion was determined to be acceptable without engineering evaluation (other than the evaluation required due to the indications of degradation in inaccessible areas – the areas subsequently examined following removal of the moisture barrier). The surface areas were accepted by examination in accordance with the provisions of IWE-3122.1. After determination that the areas were acceptable by examination, the areas of general corrosion were cleaned and the vessel was re-coated. The moisture barrier in this area was replaced on the same MAI. The NDE VT-3 report documents the re-inspection of the moisture barrier.

One area is located almost directly below the maintenance access hatch. Investigation of the area revealed that the corrosion was more extensive than originally anticipated and condition report dated 10/27/2000 was prepared to document the corrective actions associated with the corrosion on the containment vessel below the moisture barrier within the annulus region.

The Responsible Engineer (RE) provided inputs to the Owner's Activity Report (OAR-1) in response to a corrective action dated 11/12/2001. These inputs are provided in Tables 2 through 4 below:

Table 2 - Conditions in accessible areas which indicate the potential for degradation in inaccessible areas (Per 10 CFR 50.55a (b)(2)(ix)(A):

Type and Extent of Degradation	Conditions that led to degradation	Evaluation	Results of Evaluation	Necessary Corrective Action
Mechanical Damage to the inner and outer moisture barriers with some corrosion noted in 2 locations.	Wear and Tear due to traffic and work around the moisture barrier.	CR-W3-2000-1275 CA 4, Attachment 3.	During examination of the moisture barrier two areas were identified which could indicate the presence of degradation in inaccessible areas. 1) Investigation of the first area, area #13 on NDEN 200-151, revealed only limited areas of surface corrosion with no significant wall loss or pitting. All surface areas of the containment vessel at this location were determined to be acceptable by examination in accordance with IWE-3122.1. 2) Investigation of the second area, area #15 on NDEN-155, revealed more serious corrosion in the region below the moisture barrier in the annulus. CR-W3-200-1375 was prepared to document corrective actions associated with this corrosion. All surface areas examined were determined to be acceptable by examination in accordance with IWE 3122.1 following UT measurements and determination that the corrosion mechanism was not active.	1) The inner and outer moisture barriers were repaired on MAI # 421737. 2) QA NDE inspections of these areas are noted in inspection reports NDEN 2000-483 and NDEN 2000-484. 3) 100% of the moisture barrier shall be examined each refueling outage until sufficient data is obtained to allow re-evaluation by the RE to determine the optimum examination schedule. 4) Corrosion noted below the moisture barrier on the containment vessel within the annulus is considered in CR-W3-2000-1375. Area determined to be acceptable by examination in accordance with IWE 3122.1.

Table 3 - Areas with Flaws or Other Relevant Conditions Requiring Evaluation for Continued Service:

Examination Category	Item Number	Item Description	Flaw Characterization	Flaw or Relevant Condition Found During Scheduled Section XI Examination or Test? (Yes/No)
No Areas required evaluation for continued service.	N/A	N/A	N/A	N/A

Table 4 - Areas Requiring Repair, Replacement or Corrective Measures for Continued Service:

Code Class	Repair, Replaceme nt or Corrective Measure	Item Description	Description of Work	Flaw or Relevant Condition Found During Scheduled Section XI Examination or Test? (Yes/No)	Date Completed	Repair/Replacement Plan Number
MC	Repair	Moisture Barrier MB-02 Mechanical Damage in two locations.	Repair moisture barrier[1].	Yes	11/6/00	MAI 421737 Exempt from repair/replacement rules of IWA 4000 by IWA 4111
MC	Repair	Moisture Barrier MB-04 Mechanical Damage in two locations.	Repair moisture barrier[1].	Yes	11/6/00	MAI 421737 Exempt from repair/replacement rules of IWA 4000 by IWA 4111
MC	Repair	Moisture Barrier MB-05 Mechanical Damage in one location.	Repair moisture barrier[1].	Yes	11/6/00	MAI 421737 Exempt from repair/replacement rules of IWA 4000 by IWA 4111
MC	Repair	Moisture Barrier MB-06 Mechanical Damage in two locations.	Repair moisture barrier[1].	Yes	11/6/00	MAI 421737 Exempt from repair/replacement rules of IWA 4000 by IWA 4111

Code Class	Repair, Replacement or Corrective Measure	Item Description	Description of Work	Flaw or Relevant Condition Found During Scheduled Section XI Examination or Test? (Yes/No)	Date Completed	Repair/Replacement Plan Number
MC	Repair	Moisture Barrier MB-07 Mechanical Damage in two locations.	Repair moisture barrier[1].	Yes	11/6/00	MAI 421737 Exempt from repair/replacement rules of IWA 4000 by IWA 4111
MC	Repair	Moisture Barrier MB-08 Mechanical Damage in 6 locations.	Repair moisture barrier[1].	Yes	11/6/00	MAI 421737 Exempt from repair/replacement rules of IWA 4000 by IWA 4111
MC	Repair	Moisture Barrier MB-09 Mechanical Damage in one location.	Repair moisture barrier[1].	Yes	11/6/00	MAI 421737 Exempt from repair/replacement rules of IWA 4000 by IWA 4111
MC	Repair	Moisture Barrier MB-10 Mechanical Damage in 3 locations.	Repair moisture barrier[1].	Yes	11/6/00	MAI 421737 Exempt from repair/replacement rules of IWA 4000 by IWA 4111
MC	Repair	Moisture Barrier MB-11 Mechanical Damage in 3 locations.	Repair moisture barrier[1].	Yes	11/6/00	MAI 421737 Exempt from repair/replacement rules of IWA 4000 by IWA 4111
MC	Repair	Moisture Barrier MB-13 Mechanical Damage in 2 locations that overlap with MB-14.	Repair moisture barrier[1].	Yes	11/6/00	MAI 421737 Exempt from repair/replacement rules of IWA 4000 by IWA 4111

Code Class	Repair, Replacement or Corrective Measure	Item Description	Description of Work	Flaw or Relevant Condition Found During Scheduled Section XI Examination or Test? (Yes/No)	Date Completed	Repair/Replacement Plan Number
MC	Repair	Moisture Barrier MB-14 Mechanical Damage in 10 locations. (2 overlap with MB-13, 3 overlap with MB-15)	Repair moisture barrier[1].	Yes	11/6/00	MAI 421737 Exempt from repair/replacement rules of IWA 4000 by IWA 4111
MC	Repair	Moisture Barrier MB-15 Mechanical Damage in 14 locations. (3 overlap with MB-14)	Repair moisture barrier[1].	Yes	11/6/00	MAI 421737 Exempt from repair/replacement rules of IWA 4000 by IWA 4111

Note 1: Repair of moisture barriers consisted of removal of damaged areas of the moisture barrier seal

<u>RAI #7</u>

Section 9.2.3.2 of NEI 94-01, Revision 2-A, “Industry Guideline for Implementing Performance−Based Option of 10 CFR Part 50, Appendix J,” and Condition 2 in Section 4.1 of the NRC safety evaluation for NEI 94-01, Revision 2, require supplemental general visual inspections of accessible interior and exterior surfaces of the containment for structural deterioration that may affect the containment leak-tight integrity. These inspections must be 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.

Provide a schedule for a typical 15-year interval (between the last Type A test in 2005 and the proposed next Type A test in 2020), in a tabular format, of in-service inspections that were and will be performed on the containment vessel, and explain how it meets the requirements in Section 9.2.3.2 of NEI 94-01, Revision 2-A, and Condition 2 in Section 4.1 of the NRC safety evaluation NEI 94-01, Revision 2. Please include the in-service inspection intervals with the start date and end date of each inspection period, and the corresponding refueling outages.

<u>RAI #7 Response</u>

Preventative maintenance tasks exist to perform periodic general inspections of the accessible interior and exterior surfaces of the containment vessel. The table below provides a schedule for a typical 15-year interval (between the last Type A test in 2005 and proposed next Type A test in 2020) with the in-service inspection intervals with the start date and end date of each inspection period and the corresponding refueling outages.

Containment Examination Schedule

Examination Type	ISI Inspection Interval	ISI Inspection Period	Refuel Outage / Date
ILRT Type A Test	2nd Interval	3rd Period	RF13 / 2005
IWE Containment Surface Area Inspections	2nd Interval	3rd Period	RF13 / 2005
IWE Inner/Outer Moisture Barrier Inspection	2nd Interval	3rd Period	RF13 / 2005
IWE Inner/Outer Moisture Barrier Inspection	2nd Interval	3rd Period	RF14 / 2006
IWE Inner/Outer Moisture Barrier Inspection	2nd Interval	3rd Period	RF15 / 2008
IWE Inner/Outer Moisture Barrier Inspection	3rd Interval	1st Period	RF16 / 2009
IWE Containment Surface Area Inspections	3rd Interval	1st Period	RF16 / 2009
IWE Containment Bolted Connections	3rd Interval	1st Period	RF17 / 2011

IWE Containment Surface Area Inspections	3rd Interval	2nd Period	RF18 / 2012-2013
IWE Inner/Outer Moisture Barrier Inspection	3rd Interval	2nd Period	RF18 / 2012-2013
IWE Inner/Outer Moisture Barrier Inspection	3rd Interval	2nd Period	RF19 / 2014
IWE Containment Bolted Connections	3rd Interval	2nd Period	RF19 / 2014
IWE Containment Surface Area Inspections	3rd Interval	3rd Period	RF20 / 2015
IWE Inner/Outer Moisture Barrier Inspection	3rd Interval	3rd Period	RF20 / 2015
IWE Inner/Outer Moisture Barrier Inspection	3rd Interval	3rd Period	RF21 / 2017
IWE Containment Bolted Connections	3rd Interval	3rd Period	RF21 / 2017
IWE Inner/Outer Moisture Barrier Inspection	4th Interval	1st Period	RF23 / 2020
IWE Containment Bolted Connections	4th Interval	1st Period	RF23 / 2020
ILRT Type A Test	4th Interval	1st Period	RF23 / 2020

RAI #8

The LAR states that WF3 has three periods during each 10-year in-service inspection interval. Table 4-2 of the LAR presents the ASME BPV Code, Section XI, Subsection IWE, inspection results from 2003 to 2014. Please provide the following:

a. The edition of the ASME BPV Code associated with each WF3 in-service inspection interval.

b. It is not clear from the review of Table 4-2 of the LAR that 100 percent of the containment vessel accessible surface areas and the interior and exterior moisture barriers have been inspected since 2005. Please clarify or supplement the information in Table 4-2 to demonstrate that the requirements of Table IWE-2500-1 of the ASME BPV Code have been satisfied.

RAI #8a Response

Initial Interval - Containment ISI Code of Record: ASME BPV Code, Section XI, 1992 Edition with 1992 Addenda.

Second Interval – Containment ISI Code of Record: ASME BPV Code, Section XI, 1992 Edition with 1992 Addenda and ASME Section XI, 1998 Edition with 1999 and 2000 Addenda. Where subsection IWA is referenced, the 1992 Edition with the 1992 Addenda apply. Those portions of the program affected by request CEP-IWE/IWL-001 are developed in accordance with the requirements of ASME Section XI, 1998 Edition with 1999 and 2000 Addenda.

Third Interval – Containment ISI Code of Record: ASME BPV Code, Section XI, 2001 Edition with 2003 Addenda.

RAI#8b Response

The following supplemental information is added to Table 4-2 to demonstrate that the requirements of Table IWE-2500-1 of the ASME BPV Code have been satisfied.

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.

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.

The inner and outer moisture barrier inspections were performed in RF13. Inner moisture barrier sections MB-01 thru MB-12 were inspected. Six (6) areas were found to be unsatisfactory and were repaired and re-inspected with satisfactory results. Outer moisture barrier sections MB-13 through MB-15 were inspected with pitting noted in the NDE visual inspection report. The condition was accepted by the Responsible Engineer (RE) since it was a pre-existing condition which was previously identified and evaluated under a previous condition report dated 10/27/2000 and subsequently rediscovered. All areas were greater than design except one which was within design allowable tolerances. The areas were repaired and re-inspected with satisfactory results.

Fall 2006 Eleven (11) bolted connection inspections were performed in RF14 with satisfactory results. The inner and outer moisture barrier sections MB-01 through MB-15 were inspected in RF14. Inner moisture barrier sections MB-01 through MB-12 were satisfactory with no reportable damage. Pitting was noted on outer moisture barrier sections MB-13, MB-14, and MB-15 on the NDE visual examination report. The condition was accepted by the Responsible Engineer (RE) since it was a pre-existing condition which was previously identified and evaluated under a previous condition report dated 10/27/2000 and subsequently rediscovered.

Spring 2008 The inner and outer moisture barrier sections MB-01 through MB-15 were inspected in RF15. 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. The repair was performed and the condition was captured in condition report dated 5/1/2008.

The inspections performed in May 2005 (RF13), Fall 2006 (RF14), and Spring 2008 (RF15) satisfy the requirements of Table IWE-2500-1 of the ASME BPV Code for the 3rd period of the 2nd Interval.

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 RF 16 with satisfactory results.

Inside Liner Plate: In accordance with ASME Section XI Subsection IWE, a general visual inspection was performed. The examination of the liner plate met the screening criteria or was accepted by the responsible Engineer. The visual inspection was performed in accordance with the program plan and under the RE's direction. The results of this General Visual inspection reflect compliance with the building structural integrity requirements.

Annulus: All accessible areas of the outer liner plate and inner shield building were examined from the annulus area, 360° from the -1.50 ft. elevation and accessible areas from the three permanent ladders located at AZ-310, AZ-196, and AZ-133. The permanent ladder at AZ-310 goes from elevation +20 to the top of the dome. The steel liner plate was inspected in all areas – no discrepancies were found.
Note: A general inspection was performed on the liner plate surfaces required by ASME Section XI, Subsection IWE.

April 2011 Twenty-seven (27) program bolted connections were examined in RF17 with satisfactory results.

The inspections performed in November 2009 (RF16) and April 2011 (RF17) satisfy the requirements of Table IWE-2500-1 of the ASME BPV Code for the 1st period of the 3rd Interval.

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.

The inspections performed in December 2012 (RF18) and May 2014 (RF19) satisfy the requirements of Table IWE-2500-1 of the ASME BPV Code for the 2rd period of the 3nd Interval.

RAI #9

Attachment 4 of the LAR states that Table 4-1 presents summaries of the results from the WF3 shield building interior and exterior structural inspections which were performed during each refueling shutdown and prior to any integrated leak test. Contrary to this statement, Section 4.3 of the LAR states that Table 4-1 presents summaries of the results from the WF3 containment building interior and exterior structural inspections which were performed every three years and the shield building inspection was performed prior to any integrated leak test. Also, the dates included in Table 4-1 do not appear to support the statement in Attachment 4 that the WF3 shield building was inspected during each refueling outage. Please provide clarification.

RAI #9 Response

The following clarification is provided. The statement in Section 4.3 of the LAR that, Table 4-1 presents summaries of the results from the WF3 containment building interior and exterior structural inspections which were performed every three years and the shield building inspection was performed prior to any integrated leak rate test, is correct. Attachment 4 of the LAR is revised to reflect the clarification. "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 prior to any integrated leak rate test."

The following information is added to Table 4-1:

September 1995	The following interior and exterior areas of the shield building were inspected with no deficiencies noted: shield building roof, exterior shield building walls to the roof, exterior surfaces in areas of the DCT-A, DCT-B, B Switchgear, +35 penetrations rooms, MSIV A, MSIV B, MSIV passage way, -4 RAB wing area, -35 RAB wing area, and +21 RAB. All accessible penetrations, CAP valves, and the top of the containment vessel were inspected inside the annulus with no structural problems observed. Interior inspections were performed on penetrations from elevations -4, +21, electrical penetrations at +35 and +46, and the containment ring header with no structural deficiencies.
March 1999	The interior and exterior portions the steel containment vessel was performed in RF9. No indications were noted which would impair the structural integrity of the containment vessel.

RAI #10

Table 4-2 of the LAR includes the results of the inspection of the containment vessel interior coating performed in 2003. Please discuss the highlights of findings from WF3 recent inspections of the containment vessel coating and actions taken to disposition them.

RAI #10 Response

The highlights of findings from recent WF3 inspections in RF18 and RF19 of the containment vessel and actions taken to disposition them are provided below.

RF18 Inspections:
Recent containment liner plate inspections performed in RF18 were documented in NDE visual examination reports and are summarized in the table below:

Component	Description	Results
DS-05	Containment Dome Outer Surface	One 10”x8”, one 3”x3”, and three 1” areas of rust at 96’ platform also 4”x12” area of rust at 85’ platform. No pitting or wall loss at any of these areas
Construction Hatch	Surface area associated with the construction hatch	Hatch had been removed and was hanging in storage rack at the time of the examination. Removed for SGRP; Satisfactory; No indications noted
DS-01	Containment Dome Inner Surface 0°-90° Az	Satisfactory; No indications noted.
DS-03	Containment Dome Inner Surface 180°-270° Az	Satisfactory; No indications noted.
DS-04	Containment Dome Inner Surface 270°-360° Az	Satisfactory; No indications noted.
DS-02	Containment Dome Inner Surface 90°-180° Az.	Satisfactory; No indications noted.
Maintenance Hatch	Surface Area Associated with the Maintenance Hatch	Satisfactory; No indications noted.
MPAL-SA	Surface Areas of the Personnel Airlock (CB MPAL0001)	Satisfactory; No indications noted.
MPEAL-SA	Surface Areas of the Personnel Emergency Escape Airlock	Satisfactory; No indications noted.
WS-01	Containment Liner Inner Surface 0°-90° Az.@ -4 El	Satisfactory; No indications noted.
WS-02	Containment Liner Inner Surface 90°-180° Az.@ -4 El	Satisfactory; No indications noted.
WS-03	Containment Liner Inner Surface 180°-270° Az.@ -4 El	Satisfactory; No indications noted.
WS-04	Containment Liner Inner Surface 270°-360° Az.@ -4 El	Satisfactory; No indications noted.
WS-05	Containment Liner Inner Surface 0°-90° Az.@+21 El	Satisfactory; No indications noted.
WS-06	Containment Liner Inner Surface 90°-180° Az.@+21 El	Satisfactory; No indications noted.

WS-07	Containment Liner Inner Surface 180°-270° Az.@+21 El	Satisfactory; No indications noted.
WS-08	Containment Liner Inner Surface 270°- 360° Az.@+21' El	Satisfactory; No indications noted.
WS-09	Containment Liner Inner Surface 0°-90° Az.@+46 El	Satisfactory; No indications noted.
WS-10	Containment Liner Inner Surface 90°-180° Az.@+46' El	Satisfactory; No indications noted.
WS-11	Containment Liner Inner Surface 180°-270° Az.@+46' El	Satisfactory; No indications noted.
WS-12	Containment Liner Inner Surface 270°-360° Az.@+46' El	Satisfactory; No indications noted.
WS-13	Containment Liner Inner Surface 352.8°-138° Az.@+46' El	Satisfactory; No indications noted.
WS-14	Containment Liner Inner Surface 138°-207° Az.@+46' El.	Satisfactory; No indications noted.
WS-15	Containment Liner Inner Surface 207°-352.8° Az.	Active corrosion noted at lug weld to containment liner adjacent to Pen.36. No pitting or wall loss in this area. Corrosion appears to be the result of condensation dripping from chill water line located above this lug.

<u>RF19 Inspections</u>:
Recent inspections of the containment vessel coatings were performed in May 2014 (RF19). The findings from these inspections are discussed below.

"Coating failures were only found on the vessel liner plates, dome, and polar crane ring girder. Mechanical damage was observed on all the components, other than the dome. Rusting of the substrate was not observed in the areas where damage of the coatings (either from coating failure or mechanical damage) was observed. The coating system that was observed failing is Carboline Carbo Zinc 11 (CZ11) primer top coated with Carboline Phenoline 305. The type of failure is splitting of the CZ11 primer, i.e. the primer splits leaving CZ11 on the substrate. This is typical failure of this coating system.
The areas of failures are shown on the attached plate identification sheets for the liner, dome, and ring girder. A breakdown of the coating failure areas are shown below in the table.

Location	Total Area
Plates 1 to 36	525.118 ft.^{2}
Plates 37 to 162, including Construction Hatch, Maintenance Hatch, Personnel Hatch, and Escape Hatch	605.504 ft.^{2}
Dome	204.9 ft.^{2}
Ring Girder	97.7 ft.^{2}
	Total Area: 1,433.22 ft.^{2}

These failures are acceptable based on the allowable failures used for the design of the safety injection sump screen per the design calculation for Debris Generation Due to LOCA within Containment for Resolution of GS-191. According to this calculation, the allowable amount of coating failures is as follows.

Location	Allowable Failed Area
Containment Vessel Dome	3,082 ft.^{2}
Containment Liner Between Elevation 112 ft. and 138 ft.	1,144 ft.^{2}
	Total Area: 4,226 ft.^{2}

The design input records document that:

"*No extra square footage will be used for failed steel coatings. The amount of failed coatings already included for the containment liner and containment dome is conservative*."

"*Every refueling outage 10% of service level 1 coatings on structural steel are inspected in accordance with Procedure NOECP-451, Conduct Engineering Inspection of Reactor Containment Building protective coatings and commitment A8350. In addition, 100% of the containment liner plates are inspected for failed coatings. These inspections usually do not identify any failed paint on the structural steel, and limited amounts on the containment liner plates. However, any failed paint is either repaired, or added into the total of already identified failed coatings. Therefore, the total of 4,226 ft^2 is considered a conservative amount."*

Since the total found failed area of 1,433.22 ft.^{2} is less than the allowable failed area, it is acceptable to have 1,433.22 ft.^{2} of failed coatings inside containment. A condition report was initiated to document the coating failures."

RAI #11

Please discuss NRC Information Notice 2014-07, "Degradation of Leak-Chase Channel Systems for Floor Welds of Metal Containment Shell and Concrete Containment Metallic Liner," as it may apply to WF3. If applicable, discuss the operating experience, inspection results, and any corrective actions taken.

RAI #11 Response

NRC Information Notice 2014-07, "Degradation of Leak-Chase Channel Systems for Floor Welds of Metal Containment Shell and Concrete Containment Metallic Liner," was addressed in condition report dated 5/27/2014 and found that WF3's containment liner is not designed with the channel system described in the Information Notice and there are no additional actions required for WF3. Specifically, "WF3 does not have any components that should be added to the Containment Inservice Inspection Program equivalent to the items discussed in IN 2014-07. There are no channels installed to encompass the welds in the ellipsoidal bottom head of the steel containment vessel with associated pressurization lines/tubing/valves. There are no additional actions to take as part of this Information Notice. This conclusion is based on a review of design basis documents and associated controlled drawings. Additionally, a walk-down was performed in RF18 to specifically look for any covers similar to the ones identified due to in this Information Notice.

RAI #12

Please provide the following information:

a. Percent of the total number of Type B tested components that are on 120-month extended performance-based test interval.

b. Percent of the total number of Type C tested components that are on 60-month extended performance-based test interval.

RAI #12a Response

Eighty-five percent (85%) of the total number of Type B tested components are on a 120-month extended performance-based test interval.

RAI #12b Response

Forty-eight percent (48%) of the total number of Type C tested components are on a 60-month extended performance-based test interval.

Attachment 2 to

W3F1-2015-0021

Internal Events PRA Peer Review – Facts and Observations (Findings Only)

Internal Events PRA Peer Review – Facts and Observations (Findings Only)

Finding	Topic (& Associated SR)	Status	Finding/Observation	Disposition
AS-A7-01	Accident Sequence Modeling	Open for Internal Events Minimal Impact on RI-ILRT	Based on review of the WF-3 event trees and top logic model, accident sequences are not delineated for all possible scenarios - particularly in cases where a mitigating function may have succeeded. Specifically: station blackout sequences after successful power recovery, and transient sequences with successful operation of RCS pressure control. In each of these cases additional mitigating systems must be questioned to determine that the sequence terminates in a safe state.	Partially ADDRESSED A review of the event trees concluded that transient sequences with successful RCS pressure control are correctly modeled. The appropriate systems required following the successful RCS pressure control have been confirmed to ensure a safe end state. Station blackout scenarios require success of Emergency Feedwater for secondary heat removal. During the most recent revision, credit for offsite power recovery was removed for scenarios involving hardware failure of all 3 EFW pumps (PSA-WF3-01-QU). Also, many of the cutsets that would be added by additional modeling in this area would be non-minimal. Resolution of this F&O however, could have a small, but potentially noticeable impact on this application.

Internal Events PRA Peer Review – Facts and Observations (Findings Only)

Finding	Topic (& Associated SR)	Status	Finding/Observation	Disposition
AS-A7-02	Modeling of ADVs for SGTR	Open for Internal Events Negligible Impact on RI-ILRT	In the Accident Sequence Notebook, assumption 2.20 reads: ...for SGTRs, failure of ADV to close after opening is not included due to block valves upstream of the ADV that could be closed by the operator. It is not clear if after not modeling the failure to ADV to close, if the closure of the block valves by the operator has been modeled. If it is not modeled, the review team believes that it should be, so as to not lose the dependency that this operator action might have on other operator actions.	PARTIALLY ADDRESSED This finding has been evaluated though it has not been explicitly closed out. The Waterford Accident Sequence analysis (PSA-WF3-01-AS) has been revised. Modeling of the atmospheric dump valves (ADVs) to close post SGTR has been added into the top logic, and the assumptions associated with not modeling this failure mode have been removed/revised. No credit is taken for an operator action to reclose the ADV. This finding has no impact on the risk impact for this LAR since credit for this operator action would only serve to reduce impact to CDF, even with the consideration of dependencies.
AS-B3-01	Environmental Effects on Containment Equipment	Closed	The AS report (PRA-W3-01-001S01 Revision 1) includes discussion of the phenomenological impacts of heating of the containment sump water (failure of HPSI recirculation due to loss of required NPSH and pump cavitation) and large containment rupture (loss of safety injection due to the rapid depressurization, flashing of hot water in the sump, and loss of net positive suction head to the HPSI pumps) that can occur due to inadequate containment heat removal. However, some events such as steamline breaks and feedwater line breaks can result in harsh environments (especially steam and high temperature) where mitigating equipment are located.	ADDRESSED WCAP-16679-P – 'Accident Sequence Phenomena' was reviewed to determine if any phenomena other than the SLB and FLB impact were not addressed in the current Waterford AS analysis. All other phenomena have been addressed in the accident sequence and the system analyses. The effects of steam line and feed line breaks are evaluated in the initiating event document.

Internal Events PRA Peer Review – Facts and Observations (Findings Only)

Finding	Topic (& Associated SR)	Status	Finding/Observation	Disposition
DA-C2-01	Use of Condition Reports for Data Collection	Closed	The method used for collecting failure data appears to be valid, but the method for collecting unavailability data is not. If unavailability data is not tracked directly by a Maintenance Rule Function, the use of the Condition Report process to identify unavailability is not valid since planned/scheduled maintenance activities and/or testing procedures that make a pump/system unavailable will not be tracked in Condition Reports unless something goes wrong during the scheduled activity.	ADDRESSED An update to the Data Report has been performed (PSA-WF3-01-DA-01). Included in the update was a review of Operator logs to identify unavailability probabilities for those systems not tracked by the System Engineers or included in the Maintenance rule Database. The updated data is included in the internal events PRA model.
DA-C6-01	Demand Based Data Assumptions	Closed	Assumptions 8, 10, and 12 violate the requirements for calculating demands based on the ASME standard. Specifically, Assumption 8 does provide a method to ensure that Post-Maintenance demands are excluded from consideration - which is a requirement of the ASME Standard, and Assumptions 10 and 12 count changing fan speeds as demands which is inconsistent with how the fans are modeled and treated in the PRA.	ADDRESSED The internal events PRA data has been updated. The assumptions listed in the finding associated with PI data collection are no longer relevant or included in the model. The update included a review of amp hours and operator logs to capture multiple starts that could be due to post-maintenance testing and exclude them.
DA-C7-01	Documentation	Closed	No review of surveillance tests or planned maintenance activities is documented. The identification of these tests and maintenance activities, and the estimation of their frequencies (based on TS requirements of "frequency of performance" requirements) is an ASME Standard requirement for DA-C7. Review or estimation of surveillance test practices is required for requirement DA-C9.	ADDRESSED The internal events PRA data has been updated. The update utilized both MR data and operator logs to collect both plant specific failure and unavailability data (PSA-WF3-01-DA-01).

Internal Events PRA Peer Review – Facts and Observations (Findings Only)

Finding	Topic (& Associated SR)	Status	Finding/Observation	Disposition
DA-C8-01	Modeling of Normally Running Equipment in Standby	Closed	The model assumes a base, normal alignment. No consideration when base operating SSC is actually in standby, or standby SSC is operating. This may have an adverse impact on supports and dependencies. Provide rational and screening as to why the time that components are in their standby (or operating) status are not included in the model, or include the standby time in the model.	ADDRESSED Plant-specific operational records were used to determine the time that components were configured in standby status. The model has been revised to use conditional probabilities, as appropriate, for systems that have both running and standby equipment associated with them.
DA-C10-01	Plant Specific Data	Closed	An assumption in the Data Report (PRA-W3-01-001S05, Rev. 1) notes that surveillance data was embedded in the PI system and the failure data. Since failure decomposition is not employed in the WF3 PRA model, surveillance tests were not separately reviewed. The component exposure is accomplished by considering the possible opportunities for component operation. The major source of raw data on equipment operation is from the WF3 PI database. The PI database uses the information from the plant computer to determine the start and stop information on a given piece of equipment. From the start and stop information, the duration or the running hours can be determined for the piece of equipment. A review of surveillance tests was not performed to determine whether all of the exposure and failure data collected was applicable to the component failure modes.	ADDRESSED The internal events PRA data has been updated. The update includes surveillance data (via operator logs) and no longer uses the PI system (PSA-WF3-01-DA-01). The current data effort fully meets the DA-C10 SR requirements.

Internal Events PRA Peer Review – Facts and Observations (Findings Only)

Finding	Topic (& Associated SR)	Status	Finding/Observation	Disposition
DA-C12-01	Collection of Unavailability Data	Closed	Assumption 7 states that system unavailability data was only available on a monthly basis (should be able to refine this based on operator logs) and that over-estimation was potential if the system outage occurred during the month the plant was in an outage, and that this overestimation was assumed to be acceptable. Some Maintenance Rule (MR) data may be collected during lower modes, but should not be included in the at-power model data.	ADDRESSED The internal events PRA data has been updated (including plant specific unavailability). The update utilized both MR data and operator logs to collect unavailability data. The current data effort fully meets the DA-C12 SR requirements (PSA-WF3-01-DA-01). Outage time was removed from the update to the maintenance unavailability.
DA-C12-02	Plant Specific Data Collection	Closed	Section 3.3.6 states that for systems that are not tracked for Unavailability by the System Engineers, the unavailability probabilities were not updated due to a lack of data. This is not acceptable since, although the data is not tracked by the System Engineers, the data does exist.	ADDRESSED The internal events PRA data has been updated (including component unavailability). Operator logs along with maintenance records were reviewed for years 2002 through 2012 to identify unavailability of major safety systems and assessed at a train level (PSA-WF3-01-DA-01). Plant specific data was unavailable for a few specific components. For these NUREG/CR-6928 data was used.
HR-A1-01	Systematic Review for Pre-Initiators	Closed	Pre-initiators are identified in SY notebook. However, there is no related test or maintenance procedure listed. There is no evidence to show that the systematic review of procedures and practices has been done.	ADDRESSED Waterford has an extensive number of pre-initiator HRAs modeled. These events cover all standby systems and trains. The updated Waterford HRA analysis includes a systematic review of procedures and practices in evaluating pre-initiators. All pre-initiators have the relevant procedures included in the development documentation (Appendix A of PSA-WF3-01-HR).

Internal Events PRA Peer Review – Facts and Observations (Findings Only)

Finding	Topic (& Associated SR)	Status	Finding/Observation	Disposition
HR-B1-01	Pre-Initiators	Closed	There is no pre-initiator identified for CCW, because of the CCW is a running system. However, the CCW system may support the safety related standby system. The path of the CCW to support this system may be failed due to pre-initiator HFE.	ADDRESSED Restoration errors of CCW to a standby system are included in the restoration of the associated standby system (PSA-WF3-01-HR). For instance CCW to the Containment Spray pumps is included in the Containment Spray restoration logic (YHF3PMPATA and YHF3PMPATB) not the CCW logic. Therefore, these restoration errors have been identified and evaluated in the current model.
HR-D1-01	Common Miscalibration Modeling	Closed	ASEP is used for both misalignment and miscalibration. Status check may take as a credit for misalignment pre-initiator. However, it is not a credit for miscalibration. The same tool is not a factor to fail the alignment, but it is an important common factor to fail multiple miscalibration.	ADDRESSED Common calibration tools were accounted for in the common miscalibration events in the staggering of the tests. Therefore, the values are considered acceptable. Additionally, a sensitivity analysis was conducted by increasing all pre-initiator HFEs by a factor of 10 (PRA-W3-01-001S13). This sensitivity showed that the Waterford CDF only increases by 28.37% with the increase in pre-initiator events of which miscalibration events are only a fraction.
HR-F2-01	Human Failure Event Cues	Closed	The cue of each HFE is not clearly addressed.	ADDRESSED For the base PRA model, the cues (i.e., annunciators, EOP/AOP entry conditions) are explicitly discussed in each operator action in the model and are documented in the operator interview sheets. All of this is clearly documented in the updated PRA HRA analysis - PRA-WF3-01-HR.

Internal Events PRA Peer Review – Facts and Observations (Findings Only)

Finding	Topic (& Associated SR)	Status	Finding/Observation	Disposition
HR-G4-01	Human Reliability Analysis Timing	Closed	As seen in the HRA spreadsheet (hfe_cp.xls), the time available to complete actions is based on a range of references including plant-specific calculations. However, in some cases unjustified and/or inaccurate assumptions were used as a basis. The event timelines in the HRA spreadsheets also do not consistently identify the specific point in time relevant indications are received. For example, the success criteria for sump recirculation require the operators to close the minimum flow valves to the RWSP within one hour (per the HPSI and SC notebooks). The operating procedure for recirculation requires these valves to be closed within 2 minutes. No justification for the one hour timing is provided and this timing is inconsistent with the 1.82 hr used in the HRA. The justification should include consideration of the quantity of water that would be diverted to the RWSP during the time frame, the habitability impacts of containment sump water being sent to the RWSP resulting in higher radiation levels in the RAB areas that are traversed by the piping to the RWSP, and the impact this has on operator recovery actions.	ADDRESSED The recent model update included an update to the event referenced in the finding (PSA-WF3-01-HR). The new time window for the action is 1 hour (60 minutes). The new value used in the internal model reflects this reduced (from 1.8 hrs) time window. Operator interviews indicate that the action will take 2 minutes (not that it is required in 2 minutes).

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Finding	Topic (& Associated SR)	Status	Finding/Observation	Disposition
HR-G6-01	Human Reliability Analysis Documentation	Closed	A review of the summary HEP list did not indicate any issues with inconsistencies between the HEPs. There was no documentation that WSES had performed an internal consistency evaluation of their post-initiator HEPs. A discussion with the analyst indicates that they did perform an internal consistency analysis considering the scenario context, plant history, procedures, operational practices, experience, and the relative difficulties of the actions and the timing. However, because no issues were found, they did not document the review. WSES does need to document that they had performed the review, describe the general process and indicate that no discrepancies had been found. A table of HEP by HEP comparisons is not needed.	ADDRESSED Section 4.1.4.1 of the updated HRA notebook (PSA-WF3-01-HR) describes the consistency review of the modeled events.

Internal Events PRA Peer Review – Facts and Observations (Findings Only)

Finding	Topic (& Associated SR)	Status	Finding/Observation	Disposition
HR-H2-01	Modeling Non-Proceduralized Actions	Closed	As documented in the HRA report (PRA-W3-01-001S03, Rev. 1), the recovery actions included in the WF-3 PSA are not explicitly directed by procedures. Although there is no procedure or training for some of these actions, discussions with operators/TSC have evidently indicated these actions would likely be pursued (although no documentation of these discussions was included in the operator interview sheets). In fact, the worksheet for one action notes the operators do not have enough training or practice to credit the action, although it is given and HEP of 0.1. There are 9 non-proceduralized operator actions modeled. A review of these non-proceduralized actions shows that: * the time available is short (EHFMANTNR) * the action is not trained or not practiced (EHFMANTNR, MHFSAIABYR) * the working environment is poor (OHFMSSGAGR, QHEFEFWSBOR) * the decision to implement is complicated (OHFMSSGAGR) * the action is complex (QHEFEFWSBOR) All of the above items contribute to there not being sufficient justification for their credit in the model.	ADDRESSED The Waterford PRA does not credit any non-proceduralized actions (actions modeled in the past are no longer in the model). This is documented in the updated HRA report, PSA-WF3-01-HR.

Internal Events PRA Peer Review – Facts and Observations (Findings Only)

Finding	Topic (& Associated SR)	Status	Finding/Observation	Disposition
IE-A6-01	Operator Insights on IE Development	Closed	There is no evidence that interviews of operators or engineers were conducted to determine scope of initiating events or if current list of initiating events is correct. MREP information is included but it is not directly applicable to cover all of the aspects that need to be considered in the initiating event scoping and if they overlooked any initiating events.	ADDRESSED Waterford evaluated each plant system with Operations Personnel to determine if a loss of a system or train would cause a plant scram or not. Since identification of initiators reviewed all generic sources and plant specific systems (PSA-WF3-01-IE), additional initiating events types are highly unlikely to be identified. This finding was properly addressed, but insufficiently documented.
IE-C6-01	Initiating Event Fault Tree Modeling	Closed	The IE report (PRA-W3-01-001S06 Rev. 2) documents IE fault tree modeling for T9 (loss of CCW), T9RCP (loss of CCW to RCPs), TIA (loss of IA), and TTCW (loss of turbine cooling water). The initiating event fault tree modeling for these systems considers multiple failures, CCF events and routine system alignments. These IE FTs exclude many failures that are included in the systems analysis (failures of valves, breakers, etc. in redundant paths to transfer open or transfer closed; component failure rates less than 1% of the pump active failures such as sensors and transmitters; and flow diversion paths).	ADDRESSED The current IE fault tree logic is more thorough than past models. The current model is documented in PSA-WF3-01-IE. The current IE fault trees include items in redundant paths (including valves and breakers).

Internal Events PRA Peer Review – Facts and Observations (Findings Only)

Finding	Topic (& Associated SR)	Status	Finding/Observation	Disposition
IE-C12-01	Interfacing System LOCA	Open for Internal Events No Impact on RI-ILRT	ISLOCA - low pressure LPSI and HPSI line contain two check valves in series. The failure rate of the check valves need to be treated as conditional, rather than independent. Additionally need to address small ruptures in the LPSI MOVs. At present only large leakage is considered.	NOT ADDRESSED This finding is associated with the inclusion of State of Knowledge Correlation. The increase in probabilities due to SOKC would be minor per WCAP-17154-P. Additionally, a review of NUREG/CR-6928 shows that small ruptures are defined as 1 to 50 gpm. Leaks of this size are not considered sufficient to meet the classification for ISLOCA. This finding has a negligible impact on the internal events and would also have negligible impact on the risk calculation for ILRT. If any increases in ISLOCA sequences occurred because of this finding, the impact would be negligible since the Class 3a and 3b EPRI release categories used to calculate the risk metrics are determined by subtracting the contribution of Class 2 and Class 8 releases (which include ISLOCAs) from the overall CDF.

Internal Events PRA Peer Review – Facts and Observations (Findings Only)

Finding	Topic (& Associated SR)	Status	Finding/Observation	Disposition
IF-B2-01	Internal Flood	Open for Internal Events Minimal Impact on RI-ILRT	Although required by this SR, no evaluation of individual component failure modes, human-induced mechanisms, or other events that could release water into the area were identified. The evaluation assumed that using a guillotine rupture was adequate to not require any specific failures or human-induced mechanisms. This does not meet the intent or specifics of this requirement. Other SRs are also potentially not met when only the use of a guillotine rupture is used. These include: (IF-B3) Waterford 3 basically characterized all flood sources as catastrophic ruptures but where there are potential spray targets they do evaluate spray impacts. Waterford characterizes the flood in terms of gpm for larger sources or as total flood capacity for smaller flood sources. Waterford does consider pressure of the flood source to a limited extent, primarily when evaluating the potential for spray impacts. However, there is no evidence that Waterford considered the temperature of the flood source beyond stating that HELB is treated elsewhere. Waterford should include some discussion of temperature in PRA-W3-01-002. (IF-D6) Section 2.0 of the Internal Flooding analysis specifically states "all causes of flooding were considered except plant-specific maintenance activities. No mention of inclusion/exclusion of generic maintenance activities was found. While Waterford discusses operator error contributions to flooding at a very high level in section 3.1.2, basically the only floods considered were catastrophic failures. The flood scenario frequencies were then quantified using generic pipe rupture data and plant-specific pipe length. The resulting low frequencies often lead to scenarios being subsumed. While the operator induced floods may be less severe, the frequencies will be higher, so they should be considered explicitly.	NOT ADDRESSED (IF-B2) Though the intent of the SR is not met, the potential increase is most likely to be bounded by the conservative assumption listed. The frequency of human-induced failures might be higher, but the situations in which they could occur is limited. Also, operator action to mitigate these events would be more reliable since the maintenance activity would bring heightened awareness to the system. However, the exclusion of such scenarios does introduce a potential non-conservatism to the flooding analysis and potential impact to this application. (IF-B3) The majority of piping in the Waterford plant that isn't addressed by High Energy Line Breaks are of a lower temperature. This is primarily a documentation issue which affects only a small contribution to the overall results. Resolution of this Finding would have minimal impact on the results of the Flooding Analysis and would also have minimal impact on this application. (IF-D6) The analysis assumed that using guillotine rupture would bound any additional contribution from human-induced failures. Also, the frequency of human-induced failures might be higher, but the situations in which they could occur is limited. The exclusion of such scenarios does introduce a potential non-conservatism to the flooding analysis and potential impact to this application.

Internal Events PRA Peer Review – Facts and Observations (Findings Only)

Finding	Topic (& Associated SR)	Status	Finding/Observation	Disposition
IF-C3c-01	Internal Flood - Lack of Engineering Documentation	Open for Internal Events No Impact on RI-ILRT	There do not appear to be any Engineering calculations available to support some of the statements or inherent assumptions made in the Internal Flooding Analysis. In particular, room dimensions and flood rates are not available to justify flood depths stated for various rooms, some zones credit "air tight" doors as being structurally sound up to a depth of 6 inches with no justification of door integrity against a static water load of this depth, "air tight" doors appear to be treated as "flood doors" with no justification as to how this was determined (normally air tight door seals are not designed to prevent water intrusion or extrusion), timing related calculations (time for flood to reach susceptible equipment, flood rates, etc.) were not included or referenced, etc. If these calculations exist, they should be either provided in appendices to the report or referenced in the appropriate sections of the report. If the calculations do not exist, they should be performed, and the statements and inherent assumptions in the analysis re-verified to ensure they reflect the results of the calculations. On page 89 of the Internal Flooding Report, within the 2nd paragraph, a statement is made that a particular door is assumed to open out, and that the flood propagation pathway will go through that door. No discussion, or calculation, is provided to justify why that particular door will open versus another of the doors from the room (there are multiple doors associated with the room). If there is no basis behind that particular door failing prior to the other doors, then an evaluation of the flooding impacts from other doors opening should be performed. On page 215, there is an un-supported assumption that drain failures have a failure probability of 0.1. Need to provide basis for this assumption.	PARTIALLY ADDRESSED The supporting calculations referenced in the internal flooding analysis (PRA-W3-01-002) were performed by a vendor for each pipe break scenario to determine the impacts. This supporting documentation was not transmitted by the vendor along with the primary calculation and thus were not available during the Peer Review. Since the Peer Review, the supporting documents have been received from the vendor. This is a documentation only issue and would not impact this application. As for the door opening assumption, the drawing (G764) referenced in the report shows that this is the only door that opens "out" of the room other than one that lead outside the building. This is only a documentation issue. In general credit for drains less than 24" in diameter was not given for flood mitigation. Also, the factor of 0.1 was only applied in one room (RAB21-221). The scenarios involved with this room have minimal impact to the results (<1.0E-11). In general this treatment is conservative and would not impact this application.

Internal Events PRA Peer Review – Facts and Observations (Findings Only)

Finding	Topic (& Associated SR)	Status	Finding/Observation	Disposition
IF-C7-01	Internal Flood - Pump House Flood	Open for Internal Events No Impact on RI-ILRT	The Fire Water pump house has been excluded from evaluation on the basis that the failure of the fire pumps will not precipitate a reactor trip and the fire protection itself is not used to mitigate any accident scenario that might lead to core damage other than those occasioned by fire. This exclusion needs to be re-visited to determine if an internal flood in the fire water pump house has the potential to initiate a flood/spray event elsewhere in the plant due to spurious fire water valve actuations (e.g. look at potential for spray/submergence on a fire water control panel to determine if it could cause spurious signals to fire water equipment in the plant resulting in a plant spray/flood event.), and if this inadvertent actuation could result in the need for a plant shutdown. If this impact has been evaluated, document it.	NOT ADDRESSED The Fire Pump control panel cannot affect suppression system actuation inside the plant as it reacts to system pressure in the main Fire Protection Water loop. The Fire Protection Main Control Panel and Local control panels control the operation of suppression within the Reactor Auxiliary building and Turbine building. Any malfunction in the Fire Pump House would not affect the main control panel and therefore could not cause a release to damage any risk significant equipment or cause a plant trip. This Finding has no impact on the quantified results of the IF analysis and therefore no impact on the ILRT application.

Internal Events PRA Peer Review – Facts and Observations (Findings Only)

Finding	Topic (& Associated SR)	Status	Finding/Observation	Disposition
IF-D5a-01	Internal Flood - Flood Initiation Frequencies	Open for Internal Events No Impact on RI-ILRT	Although Waterford calculates the initiating event frequency for each evaluated flood scenario using generic data, and the specific calculations are presented in a footnote for each scenario, a reduction factor has been inappropriately applied to component rupture failure rates. The analysis states that the generic component failure rates are obtained from EGG-SSRE-9639 (see Table 3.2.1.2 in Flood report). However, these failure rates are then reduced by an additional factor to convert them from "spray" failures to "rupture" failures. (The example provided shows a "1/27th" reduction for a 1000 gpm valve failure) The application of the reduction factor is inappropriate since the data are "rupture" rates, not "spray" rates, and the EGG-SSRE-9639 source document has already applied a 1/25 reduction factor to ensure that the rates are applicable as rupture rates. Need to use the "rupture" failure rates without applying the additional reduction factor.	NOT ADDRESSED The analysis considers such "spray" events of having flow rates up to 100 gpm. Greater rupture rates for flood and major flood are calculated using this correlation. From EGG-SSRE-9639: "It should be kept in mind that the external rupture events include any leakage greater than 50 gpm. Therefore, most of the external ruptures identified…do not involve complete pipe severance or catastrophic failure of a valve or pump body. The frequencies for such catastrophic rupture events should be lower than those presented in this report." Use of the factor is based on the Prugh report referenced and is implemented to adjust for the size of the release to be consistent with the sizes considered for the pipe failures. This Finding has not impact to this application.
IF-D7-01	Internal Flood - Excluded Scenarios	Open for Internal Events Negligible Impact on RI-ILRT	The discussion for excluding the condensate polisher building from consideration based on the assumption that the operators would bypass the condensate polisher system in the event of a rupture/leak within the building is inadequate.	NOT ADDRESSED The worst case scenario from a flood in the condensate polisher building would be a loss of main feedwater (with a plant trip) and a loss of both 480V switchgears in the building. The Fire PRA developed a scenario with these impacts which had a CCDP of 4.68E-5 (PRA-W3-05-007) which bounds the potential effects of floods in this building. While specific flood scenarios should probably be developed for this building, it is evident that the contribution to CDF would be minor as the flood frequency still needs to be considered as well. Therefore, addressing this finding has a negligible impact this application.

Internal Events PRA Peer Review – Facts and Observations (Findings Only)

Finding	Topic (& Associated SR)	Status	Finding/Observation	Disposition
IF-D7-02	Internal Flood - Incorrect Screening Method	Open for Internal Events No Impact on RI-ILRT	The Internal Flooding report is inconsistent / incorrect in its use of "subsume" versus "screen". For example, in Section 4.2.1.3, the report states that scenarios are "subsumed" but the justification for subsuming the scenarios is based on the justification for "screening" of scenarios (screening is defined in SR IF-D7).	NOT ADDRESSED This is a documentation only finding with no impact on quantified results and therefore no impact to this application
IF-E5a-01	Internal Flood – Human Reliability Analysis	Open for Internal Events Minimal Impact on RI-ILRT	For operator actions, only actions outside of the Control Room appear to have been reviewed. Also, no analysis could be found to determine if there were any "unique" (i.e. not credited in the base PRA) operator actions that should be added for internal flooding recoveries, or if the operator actions credited were modified to account for the stress level/timing differences associated with internal flooding scenarios. Of the actions credited in the base PRA model, 4 of the operator actions appear to be removed by a recovery rule file as inaccessible. However, no additional analysis was found to justify why these 4 actions were determined to be inappropriate for internal flooding recovery, or why no other human actions were impacted by the internal flooding scenarios.	NOT ADDRESSED Operator actions outside the control room are reviewed and the operator action is not credited if the flood is on the same elevation as the component being operated locally. Three additional actions were developed for specific fire scenarios (PRA-W3-01-002). The lack of development of operator actions is one of the main sources of conservatism in this analysis. Credit for defined operator actions rather than conservative assumptions related to flood isolation would serve to greatly improve the results of this analysis. Since resolution of this finding would improve the results, it does not have negative impact to this LAR.

Internal Events PRA Peer Review – Facts and Observations (Findings Only)

Finding	Topic (& Associated SR)	Status	Finding/Observation	Disposition
IF-E6-01	Internal Flood - Uncertainties	Open for Internal Events No Impact on RI-ILRT	In general, WSES3 used the standard quantification processes from section 4.5.8 of the standard. However, WSES3 did not propagate the numerical uncertainties as part of the quantification. WSES3 needs to redo the Internal Flooding Quantification and include the propagation of the numerical uncertainties and provide the mean and ERF factors for the resultant CDFs.	NOT ADDRESSED Though no formal uncertainty analysis has been performed on the internal flooding model, this analysis is based on the internal events model which did have an uncertainty analysis performed on it. The unanalyzed uncertainty associated with this finding would be due to the initiating event frequencies in the IF analysis, that is, the pipe break frequencies. The associated error factors presented in the pipe break frequency basis document (EPRI TR-1013141) are similar to those in the internal events analysis (PSA-WF3-01-IE). Also, since the IF contribution to the ILRT is included in a conservative manner, this Finding is judged to have negligible impact on this application.
LE-F1b-01	Large Early Release Frequency - Conservatism In LERF Results	Closed	Although the LERF Model Report (PRA-W3-01-001S12, Revision 1) presents the LERF contributors, there is no discussion or review of the results to indicate there was some evaluation of the significance of various conservatisms. Although Appendix F notes that the contributors have been reviewed for reasonableness and found to be typical of what might be expected, there is no documented evidence of this review.	ADDRESSED A review of the results is documented in the quantification notebook (PSA-WF3-01-QU), not the LERF analysis. Additionally, multiple cutset review meetings have been conducted to ensure the PRA model and its results reflected the plant with reasonable accuracy. These reviews looked at the dominant (top 100 cutsets), some middle cutsets, and cutsets near the truncation limit in the combined cutset file. The sequence level cutsets were also reviewed by looking at each of the individual sequence cutset files. The insights and issues identified during these reviews are provided in Appendix F of the Quantification Notebook (PSA-WF3-01-QU).

Internal Events PRA Peer Review – Facts and Observations (Findings Only)

Finding	Topic (& Associated SR)	Status	Finding/Observation	Disposition
LE-F3-01	Large Early Release Frequency - Comparison to other plants	Open for Internal Events No Impact on RI-ILRT	Tables 4.5.8-2 d and e of the ASME Standard include requirements such as documenting a review of a sample of the significant accident sequences/cutsets, comparing the overall LERF and contributors to similar plants, reviewing a sample of non-significant cutsets, identifying significant contributors (such as initiating events, equipment failures, CCFs, and HFEs), review of component importance measures, and evaluating the overall LERF uncertainty intervals. The significant LERF contributors are presented in Section 4.3 of the LERF Report (PRA-W3-01-001S12, Revision 1), a comparison to a similar plant is presented in Section 4.5, and parametric uncertainty was performed in Appendix E, but the other requirements have not been documented.	NOT ADDRESSED The finding has been partially addressed. Every element listed in the finding has not been completed and documented. No review of importance measures is documented. Besides the review of importance measures, all listed requirements are included in the current model documentation. A review of the results is documented in the quantification notebook (PSA-WF3-01-QU), not the LERF analysis. A quantitative uncertainly evaluation (using UNCERT – a Monte Carlo sampling software) was also completed to evaluate uncertainly intervals. Additionally, multiple cutset review meetings have been conducted to ensure the PSA model and its results reflected the plant with reasonable accuracy. These reviews looked at the dominant (top 100 cutsets), some middle cutsets, and cutsets near the truncation limit in the combined cutest file. The sequence level cutsets were also reviewed by looking at each of the individual sequence cutset files. The insights and issues identified during these reviews are provided in Appendix F (PSA-WF3-01-QU). The lack of a formal review of component importance measures has no impact on the ILRT LAR.

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Finding	Topic (& Associated SR)	Status	Finding/Observation	Disposition
QU-E4-01	Sources of Uncertainty	Closed	The system notebooks identify the sources of uncertainty. However, the HRA, AS, IE and success criteria notebooks do not include any qualitative discussion of uncertainty, though some of them do address the quantitative aspects of uncertainty. This will facilitate risk informed application submittals.	ADDRESSED EPRI TR-1016737, Table A-1 provides a list of 23 topics that are issues for sources of model uncertainty. This table was reviewed to determine if these issues where addressed in the quantitative sensitivity analysis or if the characterization of the event is consistent with the EPRI report. Waterford has considered all 23 of the model uncertainty issues in the base PRA model. In addition, quantitative model sensitivity analyses were performed on several of these issues. The Waterford PRA model, as constructed and documented, facilitates risk informed application submittals.
SC-A5-01	Success Criteria - Success Beyond 24 Hours	Closed	Success criteria scenarios that are longer than 24 hours are not clearly identified as the mission time extended to a "safe, stable end-state". Documentation needs to be updated to include discussion identifying those scenarios with a longer mission time.	ADDRESSED The Success Criteria document considers the extension of the mission time beyond the nominal 24 hours if the plant is not in a safe & stable condition (PSA-WF3-01-SC). While no scenarios have extended mission times, the potential was considered.
SC-B1-01	Success Criteria - Wet/Dry Fans	Closed	There is not a clear basis for the number of wet and dry towers required for either LOCA or transient success criteria. A reference needs to be provided if available. If a reference is not available, a calculation or evaluation should be performed to ensure that the success criteria are not overly bounding - especially with regards to the transient success criteria. For example, currently transient success criteria requires either 14 dry fans or 8 wet fans but does not consider combinations of dry and wet. As this is currently modeled, the requirement for 14 dry fans may be more limiting than the success criteria for LOCAs.	ADDRESSED A calculation was performed to determine the combinations of wet and dry cooling tower fans required for success under both LOCA and transient scenarios. Success criteria have been revised to reflect the results of the new analysis (PSA-WF3-01-SC-01 - Waterford Steam Electric Station DCT/WCT Success Criteria Determination).

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Finding	Topic (& Associated SR)	Status	Finding/Observation	Disposition
SC-B1-02	Success Criteria - Hydrogen Fires	Closed	GOTHIC code was used to determine room heat-up for the various rooms in the plant. For the battery room calculation, the GOTHIC code determined that room cooling was not required for battery operation. The PRA currently requires battery room cooling due to the potential for hydrogen buildup and potential ignition during non-SBO sequences. The potential buildup of hydrogen is a habitability concern in the room but will not lead to battery failure without a fire or ignition occurring. The ignition of the potential hydrogen buildup should be considered under the fire PRA evaluation, but not as part of the base PRA model.	ADDRESSED Hydrogen fires have been accounted for in the WF3 Fire PRA. The miscellaneous hydrogen fire bin (Bin 19) has been evaluated as described in the accepted methodology (NUREG/CR-6850). Hydrogen accumulation in the battery rooms was intentionally neglected in following NUREG/CR-6850 guidance. While the specific battery room scenario in the F&O could increase risk, the amount of risk increase is considered negligible when compared to the hydrogen fires related to hydrogen systems specifically addressed in the guidance. Waterford staff explicitly followed the approved guidance in analyzing hydrogen fires. This finding therefore is not applicable to either the Internal Events or Fire PRA models.

Internal Events PRA Peer Review – Facts and Observations (Findings Only)

Finding	Topic (& Associated SR)	Status	Finding/Observation	Disposition
SC-B3-01	Success Criteria - LOCA Classifications	Closed	The current success criteria for LOCAs are based on plant capabilities and system responses. Although the definitions for small, medium and large break LOCAs are reasonable based on this criteria, the specific break sizes associated with the transitions between the LOCA definitions have not been adequately justified. Currently the break sizes are based on the original IPE criteria and no thermal hydraulic analyses of the break sizes have been performed. Per the requirement, thermal hydraulic evaluations are required at a level of detail to support the definitions/break sizes so that the appropriate initiating event frequencies can be determined. Several utilities' PRAs were dramatically impacted when the MAAP code was used to determine actual break sizes and some utilities determined that an additional fourth size LOCA was required to adequately model their plant. This has the potential to dramatically impact the CDF.	ADDRESSED The Initiating Event and Success Criteria notebooks have been updated. Updated MAAP runs based on current plant parameters were used to verify/re-define LOCA break sizes. (PSA-WF3-01-IE and PSA-WF3-01-SC)

Internal Events PRA Peer Review – Facts and Observations (Findings Only)

Finding	Topic (& Associated SR)	Status	Finding/Observation	Disposition
SC-B3-02	Success Criteria - Battery Depletion	Open for Internal Events Minimal Impact on RI-ILRT	Success criteria for the battery depletion of the A and B batteries specify that the batteries will survive for 4 hours if non-essential loads are stripped within 30 minutes. The success criteria for the AB battery specify a 6 hour coping even without any load stripping. There is no discussion of the impact or battery capability for the A & B batteries if loads are not stripped. Need to provide additional information and references for the battery depletion timing. Specific items that need to be addressed include: ● Impact of operators failing to strip loads within 30 minutes ● How long will batteries last without stripping ● Impact on potential steam generator overfill once batteries are depleted (EFW AOVs fail full open upon battery depletion, but EFW steam pump still providing full flow to both steam generators) ● Separate operator actions to strip loads need to be included in the PRA model for the AB battery and the A & B batteries. Currently a single operator action (EAFSTRBATP) is used for load stripping for all the batteries with a probability of zero failure being assigned to it. This is acceptable for the AB battery since it is not dependent upon stripping, but is not acceptable for the A & B batteries since they are dependent upon stripping loads within 30 minutes. ● Provide a reference to the SBO coping analysis (ECE89-016, Rev. 3) and any other references associated with the battery depletion calculations.	NOT ADDRESSED The basis for crediting a 6 hour coping time was a study calculation developed for PRA (ER-W3-2002-0622) by removing the conservatisms from the design basis calculations. This calculation shows that with load shedding more than 6 hours is available on A, B, and AB battery loads. Without load shedding the calculation indicates that 2.5 hours would be the most limiting time for battery depletion for A and B. Currently credit is given for the time to steam generator overfill once the EFW AOVs fail open once the batteries deplete along with at least an additional hour before core damage conditions are met (PSA-WF3-01-SC, PSA-WF3-01-AS). Even if these two time periods are considered, the 6 hours given for offsite power recovery is still slightly non conservative. A human reliability analysis was performed to analyze the operator action to shed the A and B battery loads within 30 minutes and given a value of 8.4E-2 (PSA-WF3-01-HR). Although this action has not been included in the model, its inclusion would allow for 6 hours to be allowed for offsite power recovery for over 90% of the SBO cutsets. The remaining 10% however would require higher recovery values. Station Blackout sequences accounts for roughly 45% of the cutsets (PSA-WF3-01-QU). Therefore, only 4.5% of cutsets would require an increased non-recovery factor. Even if this factor was increased five fold, the impact would be less than 25% higher CDF. Resolution of this F&O could have a small, but potentially noticeable impact on this application.

Internal Events PRA Peer Review – Facts and Observations (Findings Only)

Finding	Topic (& Associated SR)	Status	Finding/Observation	Disposition
SC-B5-02	Success Criteria - Supporting Analyses	Closed	The success criteria documentation does not explicitly discuss the reasonableness and acceptability of the thermal hydraulic and supporting engineering bases used to support the success criteria. Appendix B references old industry peer reviews and past IPE evaluations. These analyses are out of date and a new comparison to current analyses needs to be conducted.	ADDRESSED A comparison of the success criteria between WF3 and ANO2 was performed to verify the reasonableness and acceptability of the thermal hydraulic analyses and supporting engineering bases. This comparison is documented in Appendix B of PSA-WF3-01-SC.
SC-C1-02	Success Criteria - Inadequate References	Closed	Throughout the document there are a number of assumptions and statements made that directly impact the success criteria but do not have any references identified to justify their bases. Querying the PRA group determined that most of the statements were based on valid references, but they were not identified in the success criteria documentation. The references need to be specifically identified and included.	ADDRESSED The Success Criteria notebook has been updated and includes a more thorough application of references.
SC-C3-01	Success Criteria - Battery Unavailability	Closed	The success criteria notebook specifies that maintenance events associated with the batteries and chargers are included in the PRA model. A review of the model indicates that the chargers have a reasonable unavailability time modeled for them, but that the batteries currently show an unavailability of zero. This should be re-evaluated since normal practices include isolation of the batteries for discharge testing at other utilities and it should be verified if the same practice is employed here.	ADDRESSED The current (updated) model includes a value for battery unavailability.

Internal Events PRA Peer Review – Facts and Observations (Findings Only)

Finding	Topic (& Associated SR)	Status	Finding/Observation	Disposition
SY-A8-01	System Modeling - Component Boundaries	Closed	Need to reference and verify that the component boundaries used match the component failure data in the Data notebook. Pay particular attention to the diesels.	ADDRESSED The Waterford Internal Events PRA model was recently updated (August 2013). The updated system notebooks reference the Entergy document PRA-ES-01-003, which defines component boundaries. The boundaries used for diesels are correct.
SY-A12b-01	System Modeling - Flow Diversion Pathways	Closed	Need to use the exclusion criteria in SY-A14 to justify excluding flow diversion pathways. Using the criteria 2 normally closed valves should be easily justified using criteria SY-A14(a). The criteria for excluding based on a 1 to 3 ratio between the primary piping and the potential diversion piping needs to be backed up by pressure differentials. This exclusion criteria is valid if the system pressures between the primary and potential diversion piping is the same or similar. If the pressure differential is high, further analysis is required to justify exclusion. Overall, the assumptions used to exclude specific types of failures needs to be reevaluated and justification provided on how the exclusion criteria is met.	ADDRESSED Flow diversion pathways were reviewed for the Waterford 3 Fire PRA to determine if additional pathways needed to be included to address potential spurious actuation opening power operated valves. As part of this review, the flow diversion pathways excluded due to the 1/3 rule were reviewed to verify that no pressure differential is present and that sufficient margin is built into the system flow. Considerations for extended time (up to 24 hours) for systems that meet the 1/3 criteria resulted in additions to the model. Flow diversion of the CCW and CCW Makeup systems could cause system failure. These failures were added to the model (both internal events and FPRA).
SY-A12b-02	System Modeling - Flow Diversion of HPSI	Closed	The flow diversion path to the SIT (CV transfers open) causing HPSI failure is inappropriate and should be removed from the model. The check valve will have primary system pressure keeping it closed. At pressures greater than the SIT injection pressure, the check valve cannot physically open and allow water into the SIT under these conditions.	ADDRESSED The logic associated with this finding is no longer in the PRA model.

Internal Events PRA Peer Review – Facts and Observations (Findings Only)

Finding	Topic (& Associated SR)	Status	Finding/Observation	Disposition
SY-A16-01	System Modeling - Logic Sequence	Closed	OHFRCPTRIP should be ANDed with loss of CCW to Seals under gate QT05	ADDRESSED OHFRCPTRIP should not be ANDed with loss of CCW to Seals under gate QT05 because that would create circular logic. QT05 is linked by logic to the failure of Sequencer A which is needed for successful restart of the CCW pumps A and AB. A detailed review of the system logic and system operation revealed that the current model logic is correct.
SY-A16-02	System Modeling - Missing HRA	Closed	Documentation states: If a loss of CCW pump occurs, the stand by pump is started. If the second CCW pump cannot be started, then the CCW headers must be split in accordance with OP-901-510. In the event that no CCW pumps are running or can be started then the following must be performed within 3 minutes: This action doesn't appear to be modeled. Text missing in notebook (after colon).	ADDRESSED The documentation has been updated and the relevant information is now included. This action is modeled in the internal events PRA.

Internal Events PRA Peer Review – Facts and Observations (Findings Only)

Finding	Topic (& Associated SR)	Status	Finding/Observation	Disposition
SY-A18a-01	System Modeling - Coincident Unavailability	Open for Internal Events No impact on ILRT	HPSI system has an installed spare that can be aligned to either system. Coincident unavailability due to maintenance for redundant equipment is possible (spare pump OOS for extended periods and could be OOS with another pump). Need to specifically address this possibility. This may also be true for charging pumps.	NOT ADDRESSED SR SY-A18 (changed to SY-A20 in latest version of the standard) states: INCLUDE events representing the simultaneous unavailability of redundant equipment when this is a result of planned activity. The Plant Specific Failure Data Development analysis (PSA-WF3-01-DA-01) documents the inclusion of all planned concurrent maintenance (including installed spares). This remains 'not addressed' due to documentation. The coincident unavailability is included in the model, however the documentation does not fully explain the process used to consider/model events. The lack of documentation has no impact on the quantified results of the ILRT.
SY-B4-01	System Modeling - Missing CCF Combinations	Closed	Common cause failure modeling of the 2/4 failure combinations needs to be reevaluated. For example, in the HPSI model check valves SI-241-244 are currently modeled with individual component failures and combinations of 3 or 4 failures. Combinations of 2 failures are excluded. This is inappropriate since a combination of 2 failures on train A combined with the break (LOCA) on train B would fail the system success. This is currently not accounted for in the modeling. Another example in SI is the modeling of the hot leg injection isolation MOVs and CVs do not include 2/4 failures. Although there is an assumption associated with this, the logic behind the assumption no longer meets the criteria for modeling common cause failures. Non-lethal common cause combinations must be included to ensure their impact associated with individual component failures is adequately addressed.	ADDRESSED Reviewed CCF modeling of all the systems and found several CCF modeling conditions which would impact fire PRA results. This led to an immediate model update. Several previously excluded non-lethal CCF combinations were added to the model. The changes and results were summarized in the Excel spreadsheet "CCF-Disposition" and incorporated in the model.

Internal Events PRA Peer Review – Facts and Observations (Findings Only)

Finding	Topic (& Associated SR)	Status	Finding/Observation	Disposition
SY-B13-01	System Modeling - Control Room HVAC	Closed	In Table A-4 of the success criteria notebook the GOTHIC code determined that control room ventilation was required; however, Section 1.8, Major Assumption specifies that control room HVAC is not included in the HVAC system model. Although it is possible to perform a plant shutdown from the remote shutdown panel, different actions and equipment are available under this scenario and it should be considered as a recovery action, not as a standard action. Loss of control room HVAC needs to be included in the PRA model with the recovery actions assigned based on plant conditions and equipment available at the remote shutdown panels during the scenarios.	ADDRESSED The latest PRA internal events model update included the addition of MCR HVAC. Failure of control room HVAC is included in the MCR HVAC system model and notebook, and not as a sub-system in the HVAC notebook. Loss of control room HVAC is also included, in the reactor trip initiating event frequency since it is a short duration shutdown limiting condition for operation.
SY-B16-01	System Modeling - Operator Interface Dependencies	Closed	No discussion of operator interface dependencies across systems or trains are provided within the system notebooks. Need to add this discussion, or state that this information is provided in the HRA documentation (as appropriate).	ADDRESSED As part of the system model and notebook update effort, a cross-reference between the system notebooks and the HRA evaluation was added to specify that operator interface dependencies across systems or trains are addressed in the HRA documentation.
SY-C2-01	System Modeling - Intersystem CCF	Open for Internal Events No Impact on RI-ILRT	Need to add a discussion of what the criteria for CCF considerations are (which types of components were looked at, were inter- and intra-system CCFs considered, etc. If the component types were determined based off of a list from a Reference, provide this information and a pointer to the reference document/methodology.	NOT ADDRESSED This finding is a documentation issue only. During the disposition of the F&O for SY-B4, the criteria for CCF considerations were reviewed. Inter-system and Intra- system CCFs considered are documented in the CCF calculation, but not explicitly in each system notebook. The lack of documentation documented in this finding has no quantitative impact on the internal events model or the results in this LAR.

Internal Events PRA Peer Review – Facts and Observations (Findings Only)

Finding	Topic (& Associated SR)	Status	Finding/Observation	Disposition
SY-C2-02	System Modeling - Temp Diesel Generators	Closed	Statement under Operator Interface says "temporary diesel generators must be manually aligned and started as part of the accident. Therefore, …" This statement implies that the TEDGs are credited via a post-initiator operator recovery action - need to clarify that the TEDGs are not credited in the Base Model, and are only used for EOOS.	ADDRESSED Assumption 29 in the updated AC power system notebook addresses this finding: "The temporary EDGs (TEDGs) are not credited in the W3 Internal Events PRA, but are retained for EOOS alignments only."
SY-C3-01	System Modeling - Assumptions	Closed	Need to review system notebooks assumptions section and remove assumptions associated with "circular logic" and how it is handled in the system fault trees. The handling of circular logic as discussed in the system notebooks does not appear to always "mesh" with how it is described in the circular logic notebook.	ADDRESSED Assumptions related to circular logic are accurately documented in the Circular Logic Analysis (PRA-W3-01-004, Rev. 0). The assumptions in this document do "mesh" with the modeled logic. A goal of the previous revision was to remove the circular logic assumptions, from the System Notebooks. Some assumptions were inadvertently left in the notebooks. The recently revised system notebooks no longer have the circular logic discussions referenced in the finding. All content in the system notebooks related to circular logic reference the Circular Logic Analysis document.

Internal Events PRA Peer Review – Facts and Observations (Findings Only)

Finding	Topic (& Associated SR)	Status	Finding/Observation	Disposition
QU-E2-01 (Originally given as Suggestion)	Assumptions and Sources of Uncertainty	Open for Internal Events No Impact on RI-ILRT	Assumptions are identified in the systems and other Notebooks. However, there is no discussion of the impact of these assumptions on the results. It is recommended that in the QU Notebook, a qualitative discussion be provided which reviews all these assumptions and identifies a set of sensitivity runs to be made to study the impact of these assumptions on the results of the PRA. There is no documentation of a systematic review of all the PRA assumptions to identify the list of sensitivity studies to be carried out. However, many sensitivity studies have been conducted. Perform and document a systematic review of PRA assumptions to identify the list of sensitivity studies to be carried out	NOT ADDRESSED The updated PRA Standard requirement was changed to include documentation of how the PRA model is affected by model uncertainties and assumptions. However, this F&O is related to model documentation and will not quantified results. Therefore, resolution of this F&O would have no impact on this application.
QU-E4-02 (Originally given as Suggestion)	Assumptions and Sources of Uncertainty	Open for Internal Events No Impact on RI-ILRT	Additional evaluation is recommended to perform a more systematic assessment the uncertainty associated with success criteria, modeling uncertainties, degree of completeness in the selection of initiating events, and possible spatial dependencies. The requirement is to DOCUMENT assumptions and sources of uncertainty, which is met. This suggestion is to systematically assess and determine the impact of the qualitative uncertainty items.	NOT ADDRESSED The updated PRA Standard requirement was changed to include documentation of how the PRA model is affected by model uncertainties and assumptions. However, this F&O is related to model documentation and will not quantified results. Therefore, resolution of this F&O would have no impact on this application.

Attachment 3 to

W3F1-2015-0021

Calculation, Waterford 3 Evaluation of Risk Significance of an ILRT Extension

☐ 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 #** **56864**	**Page 1 of** **72**

Design Basis Calc. ☐ YES ☒ NO	☒ CALCULATION ☐ EC Markup
Calculation No: ECS14-010	**Revision: 1**
Title: Waterford 3 Evaluation of Risk Significance of an ILRT Extension	**Editorial** ☐ YES ☒ NO
System(s): MISC	**Review Org (Department): DEMECH**

Safety Class: ☐ **Safety / Quality Related** ☐ **Augmented Quality Program** ☒ **Non-Safety Related**	**Component/Equipment/Structure Type/Number:**	
Document Type: B13.40		
Keywords (Description/Topical Codes): **PRA**		

REVIEWS

Name/Signature/Date Vendor Prepared (See Page 5) **Responsible Engineer**	Name/Signature/Date Mark Thigpen (See Associated EC) ☐ **Design Verifier** ☒ **Reviewer** ☐ Comments Attached	Name/Signature/Date Chris Talazac (See Associated EC) **Supervisor/Approval** ☐ Comments Attached

CALCULATION REFERENCE SHEET	CALCULATION NO: ECS14-010 REVISION: 1

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.
1	Revised to include contribution from Internal Floods in risk calculation. Additional sensitivity cases performed for updated Level 2 model. (Reference Condition Report CR-WF3-2015-2252)

Waterford 3
Evaluation of Risk Significance of an ILRT Extension

Revision 1

April 2015

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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	4/21/2015	V. Young	RS	RS	Updated to include internal flooding PRA results
2					
3					
4					
5					

Report Quality Assurance		
Attribute (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

Report Quality Assurance		
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.	Yes	Yes
Reviewer Qualification Statement 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.	Signature/ Date R. Summitt 4/21/15	
Approver Qualification Statement 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.	R. Summitt 4/21/15	

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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 components which penetrate containment. The type B test measures 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, feedwater, and other system piping 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 [2], 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 analysis also includes the PRA results taken from the WF3 internal flooding (IF) model [23].

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)	3.46E+2	3.46E+2	3.46E+2
Type A testing risk (person-rem/yr)	1.25E-2	4.14E-2	6.25E-2
% total risk (Type A / total)	0.004%	0.012%	0.018%
Type A LERF (Class 3b) (per year)	1.41E-8	4.70E-8	7.04E-8
Changes due to extension from 10 years (current)			
Δ Risk from current (Person-rem/yr)			2.01E-2
% Increase from current (Δ Risk / Total Risk)			0.006%
Δ LERF from current (per year)			2.35E-8
Δ CCFP from current			3.53E-3
Changes due to extension from 3 years (baseline)			
Δ Risk from baseline (Person-rem/yr)			4.82E-2
% Increase from baseline (Δ Risk / Total Risk)			0.014%
Δ LERF from baseline (per year)			5.64E-8
Δ CCFP from baseline			8.47E-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 2.01E-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 2.35E-8/yr.
- The change in conditional containment failure probability (CCFP) from the current ten

(10) year interval to a fifteen (15) year interval is 3.53E-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 0.006 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 0.014 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 2.35E-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 5.64E-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 3.53E-3 (0.46 percent increase) for the proposed change and 8.47E-3 (1.10 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] and the WF3 IF model [23].

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 internal flooding initiating event model was not propagated through the Level 2 model.

However, a review of the flooding cut sets indicates that the accident sequences are similar in nature to the internal transient events. The transient initiating event contribution to the release categories is approximately fifty (50) percent INTACT and fifty (50) percent LATE. Since the internal flood scenarios are similar and the flooding scenarios would not impact the core melt and containment phenomena, the same split is applied to the CDF contribution from the internal flooding initiating events. The CDF frequency is equally split between the INTACT plant damage state (PDS) category (1.24E-6/yr) and the LATE PDS category (1.24E-6/yr). 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	1.57E-6	Class 1
LERF[1]	18	5.31E-7	Class 8
SERF	9	1.76E-9	Class 6
LATE	14	4.56E-6	Class 7
Total	N/A	6.66E-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+5
2	3.76E+5
31	1.93E+5
33	3.66E+4
61	2.76E+5
64	6.06E+5
65	1.40E+6
66	2.90E+5
67	1.35E+6
68	2.72E+6
69	6.93E+5
70	2.18E+6
71	3.91E+6
72	1.56E+6
100	3.38E+6
101	4.42E+6
103	5.80E+6
104	5.46E+6
105	6.49E+6
106	8.47E+6
107	6.27E+6
136	9.00E+6

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+6
139	1.34E+7
140	8.98E+6
142	1.41E+7
143	1.09E+7
172	1.90E+7
173	1.55E+7
175	3.24E+7
176	1.94E+7
178	4.11E+7
179	3.93E+7
301	1.27E+2
302	6.18E+2
303	3.59E+3

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 Isolation 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:

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

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	1.57E-6	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]	4.57E-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.

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.

9. The WF3 IF PRA model [23] is developed separately and was not assessed using the internal events Level 1 model [16]. Based on similar CDF scenarios and the relative independence of core damage and containment phenomenology [24], the PDS distribution is based on the internal transient events; additionally, the transient initiator is the only initiator assumed for internal flooding. This assignment provides for fifty (50) percent of the IF CDF contribution (1.24E-6/yr) to be binned as INTACT, while the remaining fifty (50) percent (1.24E-6/yr) is binned as LATE.

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	1.57E-6	6.33E+3	9.92E-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)	4.57E-6	6.95E+7	3.18E+2
8	Containment bypass	4.98E-7	5.66E+7	2.82E+1
	Total	6.66E-6		3.46E+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 (6.66E-6/yr – 1.64E-8/yr – 4.98E-7/yr) = 5.66E-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 α=0.5 and β=1. This is then combined with the existing data (no Class 3b events, 217 tests) using Equation 5.

$$p_{Class\,3b} = \frac{n+\alpha}{N+\beta} = \frac{0+0.5}{217+1} = \frac{0.5}{218} = 0.0023 \qquad \text{(eq. 5)}$$

where: N is the number of tests, n is the number of events (faults) of interest, α, β 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 (6.66E-6/yr – 4.98E-7/yr – 1.64E-8/yr) = 1.41E-8/yr (eq. 6)

<u>Class 1:</u>

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)

$FREQ_{class1}$ = 1.57E-6/yr – (5.66E-8/yr + 1.41E-8/yr) = 1.50E-6/yr

<u>Class 2:</u>

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)

<u>Class 4:</u>

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.

<u>Class 5:</u>

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.

<u>Class 6:</u>

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.

$FREQ_{class7}$ = 4.57E-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	1.50E-6	6.33E+3	9.47E-3
2	Large Containment Isolation Failures	1.64E-8	2.69E+6	4.42E-2
3a	Small Isolation Failures (Liner breach)	5.66E-8	6.33E+4[2]	3.58E-3
3b	Large Isolation Failures (Liner breach)	1.41E-8	6.33E+5[3]	8.91E-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)	4.57E-6	6.95E+7	3.18E+2
8	Containment Bypass	4.98E-7	5.66E+7	2.82E+1
	Total	6.66E-6		3.46E+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:

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

Where:

$Class3a_{BASE}$ = Class 3a person-rem/yr for baseline interval = 3.58E-3 person-rem/yr

$Class3b_{BASE}$ = Class 3b person-rem/yr for baseline interval = 8.91E-3 person-rem/yr

$Total_{BASE}$ = total person-rem/yr for baseline interval = 3.46E+2 person-rem/yr

$\%Risk_{BASE}$ = [(3.58E-3 + 8.91E-3) / 3.46E+2] x 100 = **0.004 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.

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

Risk Impact due to ten (10) year Test Interval

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]	1.33E-6	6.33E+3	8.43E-3
2	Large Containment Isolation Failures	1.64E-8	2.69E+6	4.42E-2
3a	Small Isolation Failures (Liner breach)	1.89E-7	6.33E+4	1.19E-2
3b	Large Isolation Failures (Liner breach)	4.70E-8	6.33E+5	2.97E-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)	4.57E-6	6.95E+7	3.18E+2
8	Containment Bypass	4.98E-7	5.66E+7	2.82E+1
	Total	6.66E-6		3.46E+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:

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

Where:

$Class3a_{10}$ = Class 3a person-rem/yr for current 10-year interval = 1.19E-2 person-rem/yr

$Class3b_{10}$ = Class 3b person-rem/yr for current 10-year interval = 2.97E-2 person-rem/yr

$Total_{10}$ = total person-rem/yr for current 10-year interval = 3.46E+2 person-rem/yr

$\%Risk_{10}$ = [(1.19E-2 + 2.97E-2) / 3.46E+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:

$\Delta\%Risk_{10}$ = [((Class1$_{10}$ + Class3a$_{10}$-+ Class3b$_{10}$) – (Class1$_{BASE}$ + Class3a$_{BASE}$ + Class3b$_{BASE}$))/ Total$_{BASE}$] x 100.0 (eq. 18)

Where:

Class1$_{10}$ = Class 1 person-rem/yr for current 10-year interval = 8.43E-3 person-rem/yr

Class3a$_{10}$ = Class 3a person-rem/yr for current 10-year interval = 1.19E-2 person-rem/yr

Class3b$_{10}$ = Class 3b person-rem/yr for current 10-year interval = 2.97E-2 person-rem/yr

Class1$_{BASE}$ = Class 1 person-rem/yr for baseline interval = 9.47E-3 person-rem/yr (Table 12)

Class3a$_{BASE}$ = Class 3a person-rem/yr for baseline interval = 3.58E-3 person-rem/yr (Table 12)

Class3b$_{BASE}$ = Class 3b person-rem/yr for baseline interval = 8.91E-3 person-rem/yr (Table 12)

Total$_{BASE}$ = total person-rem/yr for baseline interval = 3.46E+2 person-rem/yr (Table 12)

$\Delta\%Risk_{10}$ = [(8.43E-3 + 1.19E-2 + 2.97E-2) – (9.47E-3 + 3.58E-3 + 8.91E-3)] / 3.46E+2 x 100.0 = **0.008 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.

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

<u>Risk Impact due to 15-year Test Interval</u>

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]	1.21E-6	6.33E+3	7.68E-3
2	Large Containment Isolation Failures	1.64E-8	2.69E+6	4.42E-2
3a	Small Isolation Failures (Liner breach)	2.83E-7	6.33E+4	1.79E-2
3b	Large Isolation Failures (Liner breach)	7.04E-8	6.33E+5	4.46E-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)	4.57E-6	6.95E+7	3.18E+2
8	Containment Bypass	4.98E-7	5.66E+7	2.82E+1
	Total	6.66E-6		3.46E+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 = 1.79E-2 person-rem/yr

$Class3b_{15}$ = Class 3b person-rem/yr for 15-year interval = 4.46E-2 person-rem/yr

$Total_{15}$ = total person-rem year for 15-year interval = 3.46E+2 person-rem/yr

$\%Risk_{15}$ = [(1.79E-2 + 4.46E-2) / 3.46E+2] x 100 = **0.018 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 = 7.68E-3 person-rem/yr

$Class3a_{15}$ = Class 3a person-rem/yr for current 15-year interval = 1.79E-2 person-rem/yr

$Class3b_{15}$ = Class 3b person-rem/yr for current 15-year interval = 4.46E-2 person-rem/yr

$Class1_{BASE}$ = Class 1 person-rem/yr for baseline interval = 9.47E-3 person-rem/yr (Table 12)

$Class3a_{BASE}$ = Class 3a person-rem/yr for baseline interval = 3.58E-3 person-rem/yr (Table 12)

$Class3b_{BASE}$ = Class 3b person-rem/yr for baseline interval = 8.91E-3 person-rem/yr (Table 12)

$Total_{BASE}$ = total person-rem/yr for baseline interval = 3.46E+2 person-rem/yr (Table 12)

$\Delta\%Risk_{15}$ = [(7.68E-3 + 1.79E-2 + 4.46E-2) – (9.47E-3 + 3.58E-3 + 8.91E-3)] / 3.46E+2 x 100.0 = **0.014 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 = 7.68E-3 person-rem/yr

$Class3a_{15}$ = Class 3a person-rem/yr for current 15-year interval = 1.79E-2 person-rem/yr

$Class3b_{15}$ = Class 3b person-rem/yr for current 15-year interval = 4.46E-2 person-rem/yr

$Class1_{10}$ = Class 1 person-rem/yr for current 10-year interval = 8.43E-3 person-rem/yr (Table 13)

$Class3a_{10}$ = Class 3a person-rem/yr for current 10-year interval = 1.19E-2 person-rem/yr (Table 13)

$Class3b_{10}$ = Class 3b person-rem/yr for current 10-year interval = 2.97E-2 person-rem/yr (Table 13)

$Total_{10}$ = total person-rem/yr for 10-year interval = 3.46E+2 person-rem/yr (Table 13)

% $Total_{10-15}$ = [(7.68E-3 + 1.79E-2 + 4.46E-2) – (8.43E-3 + 1.19E-2 + 2.97E-2)] / 3.46E+2 x 100 = **0.006 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)	1.41E-8/yr	4.70E-8/yr	7.04E-8/yr
ΔLERF (3 year baseline)		3.29E-8/yr	5.64E-8/yr
ΔLERF (10 year baseline)			2.35E-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 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 2.35E-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 5.64E-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)	1.55E-6	1.52E-6	1.50E-6
f(ncf)/CDF	2.33E-1	2.29E-1	2.25E-1
CCFP	7.67E-1	7.71E-1	7.75E-1
ΔCCFP (3 year baseline)		4.94E-3	8.47E-3
ΔCCFP (10 year baseline)			3.53E-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 3.53E-3 or 0.46 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 8.47E-3 or 1.10 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 31st 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

<table>
<tr><th>Step</th><th>Description</th><th colspan="2">Containment Cylinder and Dome (85%)</th><th colspan="2">Containment Basemat (15%)</th></tr>
<tr><td>1</td><td>Historical liner flaw likelihood
Failure data: containment location specific
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</td><td colspan="2">Events 6
6 / (66 x 17.25) = 5.27E-3/yr</td><td colspan="2">Events: 0
Assume a half failure
0.5 / (66 x 17.25) = 4.39E-4/yr</td></tr>
<tr><td rowspan="2">2</td><td rowspan="2">Aged adjusted liner flaw likelihood
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.</td><td>Year
1
average 5-10
15</td><td>Failure rate
2.14E-3/yr
5.27E-3/yr
1.49E-2/yr</td><td>Year
1
average 5-10
15</td><td>Failure rate
1.78E-4/yr
4.39E-4/yr
1.24E-3/yr</td></tr>
<tr><td colspan="2">15 year average = 6.42E-3/yr</td><td colspan="2">15 year average = 5.58E-4/yr</td></tr>
<tr><td>3</td><td>Increase in flaw likelihood between 3 and 15 years
Uses aged adjusted liner flaw likelihood (Step 2), assuming failure rate doubles every five years.</td><td colspan="2">0.74% (1 to 3 years)
4.24% (1 to 10 years)
9.63% (1 to 15 years)</td><td colspan="2">0.06% (1 to 3 years)
0.36% (1 to 10 years)
0.84% (1 to 15 years)</td></tr>
<tr><td>4</td><td>Likelihood of breach in containment given liner flaw</td><td colspan="2">1%</td><td colspan="2">0.1%</td></tr>
</table>

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% 5% failure to identify visual flaws plus 5% likelihood that the flaw is not visible (not through-cylinder but could be detected by ILRT) All events have been detected through visual inspection. 5% visible failure detection is a conservative assumption.	100% Cannot be visually inspected
6	Likelihood of non-detected containment leakage (Steps 3 x 4 x 5)	0.00074% (3 years) 0.74% x 1% x 10% 0.00424% (10 years) 4.24% x 1% x 10% 0.00963% (15 years) 9.63% x 1% x 10%	0.00006% (3 years) 0.06% x 0.1% x 100% 0.00036% (10 years) 0.36% x 0.1% x 100% 0.00084% (15 years) 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	6.66E-6	1.64E-8	4.98E-7	1.41E-8	8.00E-6	4.90E-11
10-years	6.66E-6	1.64E-8	4.98E-7	4.70E-8	4.60E-5	2.80E-10
15-years	6.66E-6	1.64E-8	4.98E-7	7.04E-8	1.05E-4	6.36E-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	1.50E-6	9.47E-3	1.50E-6	9.47E-3	-3.10E-7	1.33E-6	8.43E-3	1.33E-6	8.43E-3	-1.77E-6	1.21E-6	7.68E-3	1.21E-6	7.69E-3	-4.02E-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	5.66E-8	3.58E-3	5.66E-8	3.58E-3	N/A	1.89E-7	1.19E-2	1.89E-7	1.19E-2	N/A	2.83E-7	1.79E-2	2.83E-7	1.79E-2	N/A
3b	1.41E-8	8.91E-3	1.41E-8	8.95E-3	3.10E-5	4.70E-8	2.97E-2	4.72E-8	2.99E-2	1.77E-4	7.04E-8	4.46E-2	7.11E-8	4.50E-2	4.02E-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	4.57E-6	3.18E+2	4.57E-6	3.18E+2	N/A	4.57E-6	3.18E+2	4.57E-6	3.18E+2	N/A	4.57E-6	3.18E+2	4.57E-6	3.18E+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	6.66E-6	3.46E+2	6.66E-6	3.46E+2	3.07E-5	6.66E-6	3.46E+2	6.66E-6	3.46E+2	1.76E-4	6.66E-6	3.46E+2	6.66E-6	3.46E+2	3.98E-4
Class 3b LERF	1.41E-8		1.41E-8 (4.90E-11)			4.70E-8		4.72E-8 (2.80E-10)			7.04E-8		7.11E-8 (6.36E-10)		
Delta LERF (from base case of 3 per 10 years)						3.29E-8		3.31E-8 (2.31E-10)			5.64E-8		5.69E-8 (5.87E-10)		
Delta LERF from 1 per 10 years						N/A					2.35E-8		2.38E-8 (3.55E-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	9.22E-3	3.88E-3	58%
100	2.29E-3	2.47E-4	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	1.57E-6	1.54E-6	6.33E+3	9.76E-3	1.48E-6	9.39E-3	1.44E-6	9.12E-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	2.39E-8	6.33E+4	1.51E-3	7.97E-8	5.04E-3	1.19E-7	7.56E-3
3b	N/A	1.52E-9	6.33E+5	9.63E-4	5.07E-9	3.21E-3	7.61E-9	4.81E-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	4.57E-6	4.57E-6	6.95E+7	3.18E+2	4.57E-6	3.18E+2	4.57E-6	3.18E+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	6.66E-6	6.66E-6	1.30E+8	3.46E+2	6.66E-6	3.46E+2	6.66E-6	3.46E+2
Δ LERF (3 per 10 yrs base)					3.55E-9		6.08E-9	
Δ LERF (1 per 10 yrs base)							2.54E-9	
CCFP	7.65E-1				7.65E-1		7.66E-1	

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	1.41E-8	4.70E-8	7.04E-8	2.35E-8
Combined	5.29E-8	1.76E-7	2.65E-7	8.82E-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

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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

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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_R1.xlsx	104.5 KB	4/20/2015

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.

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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? 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

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
Revision 1		
1. Editorial comments.	Updated report with all editorial changes.	RS

2. Section 2.0: I think it is good to be a bit more descriptive of how the 50/50 split was derived. Once that is done the Table 2 notes can be removed I believe.	Additional wording for 50/50 split description added.	RS
3. Please check the change made to Reference 4.	No change required; the EPRI report is Reference 2 (Revision 2-A of 1009325). Reference 4 is the NEI ILRT extension task force document (Revision 3-A).	RS

Attachment 1: Sensitivity Cases

CALCULATION COVER SHEET

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File Name:
RSC-CALKNX-2015-0403.docx

FORM NO.: RSC-CALC99-02 Rev 25

Calculation No:	RSC-CALKNX-2015-0403	Revision:	0
Title: ILRT Sensitivity - Summary of Risk Impact on Extending A ILRT Test Frequency Using Internal Flooding Results and Updated Level 2 Results		Facility:	Waterford Steam Electric Station Unit 3
Client:	Entergy	Project:	RSC 15 - 03

Document Control Information	
This Calculation:	Calculation Abstract and Search Keywords
☒ Contains RSC Proprietary Information ☐ Is a New Document ☐ Supports, ☐ Amends, ☐ Supersedes, ☒ Or Supplements RSC Document/Calculations(s):RSC 14-12/ECS14-010, Rev. 1	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, using the internal flooding results and updated Level 2 results. Keywords (up to 3): ILRT, WF3, Level 2

RSC Quality Assurance QUESTIONS MUST BE ANSWERED PRIOR TO INITIATING WORK REMEMBER TO STOP, THINK, ACT, REVIEW	
HAVE YOU CHECKED YOUR QUALS AND YOU QUALIFIED TO PERFORM THIS CALCULATION?	Yes
ARE YOU USING THE CURRENT RSC STANDARDS OR APPROVED CLIENT STANDARDS?	Yes
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RSC SOFTWARE/APPLICATION REFERENCE

☐ CAFTA	☐ MAAP	☐ TIFA	☐ FATCAT	☐ EOOS	☐ FDT/CFAST/FDS
☒ Office 2010	☐ FRANC/X	☐ SIP	☐ MISTKE	☐ PRAMS	☐ OTHER: ;

	Originator	Reviewer	Approver
Electronic Signature			
Name	Vincent Young	Ricky Summitt	Ricky Summitt
Date	04 / 17 / 15	04 / 20 / 15	04 / 20 / 15

RSC	**CALCULATION SHEET** **PRINTED APRIL 23, 2015** File Saved: 4/23/2015 3:56:00 PM	**Page 2 of 18**
File Name: RSC-CALKNX-2015-0403.docx		**FORM NO.: RSC-CALC99-02 Rev 25**

Calc. No.: RSC-CALKNX-2015-0403	Revision: 0
Calculation Title: ILRT Sensitivity - Summary of Risk Impact on Extending A ILRT Test Frequency Using Internal Flooding Results and Updated Level 2 Results	

PURPOSE OF ANALYSIS

This sensitivity calculation provides a summary 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), using the Level 2 probabilistic risk assessment (PRA) results from RSC 13-02/PSA-WF3-01-LE[2] and adding the internal flooding PRA results from PRA-W3-01-002[7]. A second sensitivity is also performed, using the same Level 2 PRA results while doubling the internal flooding core damage frequency (CDF) contribution from PRA-W3-01-002. 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[3]. The complete ILRT extension risk analysis is provided in RSC 14-12/ECS14-010, Rev. 1[8].

METHODOLOGY/APPROACH/PROCESS (define analysis steps)

The reactor containment leakage test program consists of three tests: Type A, Type B, and Type C (Reference 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 components which penetrate containment. The Type B test measures 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, feedwater, and other system piping 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 calculation documents a risk-based evaluation of the proposed change of the ILRT interval for the WF3, specifically using the WF3 internal flooding PRA results provided in PRA-W3-01-002 (Reference 7) and the WF3 Level 2 PRA results in RSC 13-02 (Reference 2). The proposed change would only impact testing associated with the current surveillance tests for Type A leakage, procedure PE-005-001[4].

This summary utilizes the guidelines set forth in NEI 94-01[5], the methodology used in the EPRI Report, and considers the submittals generated by other utilities.

The complete ILRT extension risk analysis is provided in RSC 14-12/ECS14-010, Rev. 1 (Reference 8).

ANALYSIS WORK AREA

Sensitivity Case #1 (Updated Level 2 Results + Internal Flooding Results) - Summary

The sensitivity results that combine the WF3 internal flooding model results (Reference 7) with the updated WF3 Level 2 PRA results (Reference 2) are provided 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. Note that for this sensitivity case, the entire CDF contribution from the internal flooding PRA model (Reference 7) is split between the INTACT plant damage state (PDS) category and the LATE PDS category; 50 percent of the contribution (1.24E-6/yr) is binned as INTACT, while the remaining 50 percent (1.24E-6/yr) is binned as LATE.

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Table 1. Sensitivity #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)	3.76E+2	3.76E+2	3.76E+2
Type A testing risk (person-rem/yr)	1.70E-2	5.68E-2	8.51E-2
% total risk (Type A / total)	0.005%	0.015%	0.023%
Type A LERF (Class 3b) (per year)	1.92E-8	6.40E-8	9.60E-8
Changes due to extension from 10 years (current)			
Δ Risk from current (Person-rem/yr)			2.74E-2
% Increase from current (Δ Risk / Total Risk)			0.007%
Δ LERF from current (per year)			3.20E-8
Δ CCFP from current			3.79E-3
Changes due to extension from 3 years (baseline)			
Δ Risk from baseline (Person-rem/yr)			6.57E-2
% Increase from baseline (Δ Risk / Total Risk)			0.017%
Δ LERF from baseline (per year)			7.68E-8
Δ CCFP from baseline			9.08E-3

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 2.74E-2 person-rem/year.

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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 3.20E-8/yr.

The change in conditional containment failure probability (CCFP) from the current ten (10) year interval to a fifteen (15) year interval is 3.79E-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 0.007 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 0.017 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 3.20E-8/yr. Guidance in Regulatory Guide 1.174 defines very small changes in LERF as below 10^{-7}/yr; therefore, increasing the ILRT interval from ten (10) to fifteen (15) years is 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 7.68E-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 3.79E-3 (0.57 percent increase) for the proposed change and 9.08E-3 (1.39 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 (Reference 3). Therefore, the changes in CCFP are considered small and demonstrate that the defense-in-depth philosophy is maintained.

In reviewing the results for sensitivity case #1, the WF3 analysis demonstrates that the change in plant risk is small as a result of this proposed extension of ILRT testing sensitivity. The change in LERF defined in the analysis for both the baseline and the current cases is within the acceptance criterion.

<u>Sensitivity Case #1 – Detailed Analysis</u>

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.

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Table 2. Release Category Frequencies

Release Category	Contributing WF3 Accident Categories	Frequency (/yr)	EPRI Classification
INTACT (S)[1]	10	2.94E-6	Class 1
LERF[2]	18	8.25E-8	Class 8
SERF	9	8.87E-8	Class 6
LATE[3]	14	5.34E-6	Class 7
Total	N/A	8.45E-6	N/A

1. 50 percent of the CDF contribution from the internal flooding PRA model [(0.5) * (2.48E-6/yr)] was binned in the INTACT release category.
2. The LERF contribution for WF3 contains early containment failures due to containment phenomenon; per 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 (Reference 3).
3. 50 percent of the CDF contribution from the internal flooding PRA model [(0.5) * (2.48E-6/yr)] was binned in the LATE release category.

Table 3 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 (Table 5 of Reference 8) 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.

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Table 3. WF3 Dose for EPRI Accident Classes

Release Category	Frequency (/yr)	EPRI Class	WF3 Dose (person-rem)
INTACT	2.94E-6	Class 1	6.33E+3
LERF[1]	5.81E-9	Class 2	2.69E+6
SERF[2]	8.87E-8	Class 6	2.69E+5[3]
LERF + LATE[4]	5.34E-6	Class 7	6.95E+7
LERF[5]	7.67E-8	Class 8	5.66E+7

1. The EPRI Class 2 category consists of the WF3 assigned LERF contribution associated with isolation failures (Table 24 of Reference 2).
2. The EPRI Class 6 category consists of WF3 assigned scrubbed isolation failures in SERF.
3. The EPRI Class 6 dose rate is derived from the Class 2 dose 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 (Table 24 of Reference 2). Per the EPRI guidance document, LATE failures are also classified as Class 7.
5. The EPRI Class 8 category consists of the WF3 assigned LERF contribution associated with bypass or SGTR failures (Table 24 of Reference 2).

Table 4 summarizes the information in Section 4.2 of Reference 8, by the EPRI-defined classes. This table also presents dose exposures 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 (Reference 3).

The Class 3a and 3b doses are represented as $10L_a$ and $100L_a$, respectively. Table 4 also presents the person-rem frequency data determined by multiplying the failure class frequency by the corresponding exposure.

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Table 4. Baseline Risk Profile

Class	Description	Frequency (/yr)	Person-rem	Person-rem (/yr)
1	No Containment Failure	2.84E-6	6.33E+3	1.80E-2
2	Large Containment Isolation Failures	5.81E-9	2.69E+6	1.56E-2
3a	Small Isolation Failures (Liner breach)	7.71E-8	6.33E+4[2]	4.88E-3
3b	Large Isolation Failures (Liner breach)	1.92E-8	6.33E+5[3]	1.21E-2
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)	8.87E-8	2.69E+5	2.38E-2
7	Severe Accident Phenomena-induced Failure (Early and Late)	5.34E-6	6.95E+7	3.71E+2
8	Containment Bypass	7.67E-8	5.66E+7	4.35E+0
	Total	8.45E-6		3.76E+2

1. Represents a probabilistically insignificant value.
2. 10 times L_a.
3. 100 times L_a.

Based on the approved EPRI methodology (Reference 3) and the NEI guidance (Reference 5), 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 5 below.

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Table 5. Risk Profile for Once in Ten Year Testing

Class	Description	Frequency (/yr)	Person-rem[2]	Person-rem (/yr)
1	No Containment Failure[1]	2.62E-6	6.33E+3	1.66E-2
2	Large Containment Isolation Failures	5.81E-9	2.69E+6	1.56E-2
3a	Small Isolation Failures (Liner breach)	2.57E-7	6.33E+4	1.63E-2
3b	Large Isolation Failures (Liner breach)	6.40E-8	6.33E+5	4.05E-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)	8.87E-8	2.69E+5	2.38E-2
7	Severe Accident Phenomena-induced Failure (Early and Late)	5.34E-6	6.95E+7	3.71E+2
8	Containment Bypass	7.67E-8	5.66E+7	4.35E+0
	Total	8.45E-6		3.76E+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 4.
3. Represents a probabilistically insignificant value.

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 6 below.

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Table 6. Risk Profile for Once in Fifteen Year Testing

Class	Description	Frequency (/yr)	Person-rem [2]	Person-rem (/yr)
1	No Containment Failure[1]	2.46E-6	6.33E+3	1.55E-2
2	Large Containment Isolation Failures	5.81E-9	2.69E+6	1.56E-2
3a	Small Isolation Failures (Liner breach)	3.86E-7	6.33E+4	2.44E-2
3b	Large Isolation Failures (Liner breach)	9.60E-8	6.33E+5	6.07E-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)	8.87E-8	2.69E+5	2.38E-2
7	Severe Accident Phenomena-induced Failure (Early and Late)	5.34E-6	6.95E+7	3.71E+2
8	Containment Bypass	7.67E-8	5.66E+7	4.35E+0
	Total	8.45E-6		3.76E+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 4.
3. Represents a probabilistically insignificant value.

Sensitivity Case #2 (Updated Level 2 Results + Doubled Internal Flooding Results) - Summary

The sensitivity results that combine the WF3 internal flooding model results (Reference 7) with the updated WF3 Level 2 PRA results (Reference 2) are provided in Table 7 below. Type A testing risk is comprised of EPRI Class 3a and Class 3b. Class 3b is defined as the LERF contribution to Type A testing. Note that for this sensitivity case, the entire CDF contribution from the internal flooding PRA model (Reference 7) is doubled, then split between the INTACT plant damage state (PDS) category and the LATE PDS category; 50 percent (2.48E-6/yr) is binned as INTACT, while the remaining 50 percent (2.48E-6/yr) is binned as LATE.

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Table 7. 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)	4.62E+2	4.62E+2	4.62E+2
Type A testing risk (person-rem/yr)	2.21E-2	7.36E-2	1.10E-1
% total risk (Type A / total)	0.005%	0.016%	0.024%
Type A LERF (Class 3b) (per year)	2.49E-8	8.29E-8	1.24E-7
Changes due to extension from 10 years (current)			
Δ Risk from current (Person-rem/yr)			3.55E-2
% Increase from current (Δ Risk / Total Risk)			0.008%
Δ LERF from current (per year)			4.15E-8
Δ CCFP from current			3.79E-3
Changes due to extension from 3 years (baseline)			
Δ Risk from baseline (Person-rem/yr)			8.51E-2
% Increase from baseline (Δ Risk / Total Risk)			0.018%
Δ LERF from baseline (per year)			9.95E-8
Δ CCFP from baseline			9.11E-3

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 3.55E-2 person-rem/year.

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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 4.15E-8/yr.

The change in conditional containment failure probability (CCFP) from the current ten (10) year interval to a fifteen (15) year interval is 3.79E-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 0.008 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 0.018 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 4.15E-8/yr. Guidance in Regulatory Guide 1.174 defines very small changes in LERF as below 10^{-7}/yr; therefore, increasing the ILRT interval from ten (10) to fifteen (15) years is 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 9.95E-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 3.79E-3 (0.61 percent increase) for the proposed change and 9.11E-3 (1.47 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 (Reference 3). Therefore, the changes in CCFP are considered small and demonstrate that the defense-in-depth philosophy is maintained.

In reviewing the results for sensitivity case #2, the WF3 analysis demonstrates that the change in plant risk is small as a result of this proposed extension of ILRT testing sensitivity. The change in LERF defined in the analysis for both the baseline and the current cases is within the acceptance criterion.

Sensitivity Case #2 – Detailed Analysis

The WF3 release states are summarized in Table 8. 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 8.

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Table 8. Release Category Frequencies

Release Category	Contributing WF3 Accident Categories	Frequency (/yr)	EPRI Classification
INTACT (S)[1]	10	4.18E-6	Class 1
LERF[2]	18	8.25E-8	Class 8
SERF	9	8.87E-8	Class 6
LATE[3]	14	6.58E-6	Class 7
Total	N/A	1.09E-5	N/A

1. 50 percent of the CDF contribution from the internal flooding PRA model was doubled [(0.5) * (2.48E-6/yr) * (2)] and binned in the INTACT release category.
2. The LERF contribution for WF3 contains early containment failures due to containment phenomenon; per 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 (Reference 3).
3. 50 percent of the CDF contribution from the internal flooding PRA model was doubled [(0.5) * (2.48E-6/yr) * (2)] and binned in the LATE release category.

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 (Table 5 of Reference 8) 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.

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Table 9. WF3 Dose for EPRI Accident Classes

Release Category	Frequency (/yr)	EPRI Class	WF3 Dose (person-rem)
INTACT	4.18E-6	Class 1	6.33E+3
LERF[1]	5.81E-9	Class 2	2.69E+6
SERF[2]	8.87E-8	Class 6	2.69E+5[3]
LERF + LATE[4]	6.58E-6	Class 7	6.95E+7
LERF[5]	7.67E-8	Class 8	5.66E+7

1. The EPRI Class 2 category consists of the WF3 assigned LERF contribution associated with isolation failures (Table 24 of Reference 2).
2. The EPRI Class 6 category consists of WF3 assigned scrubbed isolation failures in SERF.
3. The EPRI Class 6 dose rate is derived from the Class 2 dose 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 (Table 24 of Reference 2). Per the EPRI guidance document, LATE failures are also classified as Class 7.
5. The EPRI Class 8 category consists of the WF3 assigned LERF contribution associated with bypass or SGTR failures (Table 24 of Reference 2).

Table 10 summarizes the information in Section 4.2 of Reference 8, by the EPRI-defined classes. This table also presents dose exposures 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 (Reference 3).

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

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Table 10. Baseline Risk Profile

Class	Description	Frequency (/yr)	Person-rem	Person-rem (/yr)
1	No Containment Failure	4.06E-6	6.33E+3	2.57E-2
2	Large Containment Isolation Failures	5.81E-9	2.69E+6	1.56E-2
3a	Small Isolation Failures (Liner breach)	1.00E-7	6.33E+4[2]	6.33E-3
3b	Large Isolation Failures (Liner breach)	2.49E-8	6.33E+5[3]	1.57E-2
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)	8.87E-8	2.69E+5	2.38E-2
7	Severe Accident Phenomena-induced Failure (Early and Late)	6.58E-6	6.95E+7	4.58E+2
8	Containment Bypass	7.67E-8	5.66E+7	4.35E+0
	Total	1.09E-5		4.62E+2

1. Represents a probabilistically insignificant value.
2. 10 times L_a.
3. 100 times L_a.

Based on the approved EPRI methodology (Reference 3) and the NEI guidance (Reference 5), 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 11 below.

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Table 11. Risk Profile for Once in Ten Year Testing

Class	Description	Frequency (/yr)	Person-rem[2]	Person-rem (/yr)
1	No Containment Failure[1]	3.76E-6	6.33E+3	2.38E-2
2	Large Containment Isolation Failures	5.81E-9	2.69E+6	1.56E-2
3a	Small Isolation Failures (Liner breach)	3.33E-7	6.33E+4	2.11E-2
3b	Large Isolation Failures (Liner breach)	8.29E-8	6.33E+5	5.25E-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)	8.87E-8	2.69E+5	2.38E-2
7	Severe Accident Phenomena-induced Failure (Early and Late)	6.58E-6	6.95E+7	4.58E+2
8	Containment Bypass	7.67E-8	5.66E+7	4.35E+0
	Total	1.09E-5		4.62E+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 10.
3. Represents a probabilistically insignificant value.

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 12 below.

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Table 12. Risk Profile for Once in Fifteen Year Testing

Class	Description	Frequency (/yr)	Person-rem [2]	Person-rem (/yr)
1	No Containment Failure[1]	3.56E-6	6.33E+3	2.25E-2
2	Large Containment Isolation Failures	5.81E-9	2.69E+6	1.56E-2
3a	Small Isolation Failures (Liner breach)	5.00E-7	6.33E+4	3.16E-2
3b	Large Isolation Failures (Liner breach)	1.24E-7	6.33E+5	7.87E-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)	8.87E-8	2.69E+5	2.38E-2
7	Severe Accident Phenomena-induced Failure (Early and Late)	6.58E-6	6.95E+7	4.58E+2
8	Containment Bypass	7.67E-8	5.66E+7	4.35E+0
	Total	1.09E-5		4.62E+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 10.
3. Represents a probabilistically insignificant value.

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SCANS OR ATTACHMENTS (as necessary)

Attachment 4 to

W3F1-2015-0021

Revised Section 4.5.3 of License Amendment Request

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 2.01E-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 2.35E-8/yr.

- The change in conditional containment failure probability (CCFP) from the current ten (10) year interval to a fifteen (15) year interval is 3.53E-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 0.006 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 0.014 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 2.35E-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 5.64E-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 3.53E-3 (0.46 percent increase) for the proposed change and 8.47E-3 (1.10 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.