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10 CFR 50.90

W3F1-2021-0050

October 1, 2021

ATTN: Document Control Desk
U.S. Nuclear Regulatory Commission
Washington, DC 20555-0001

Subject: Response to U. S. Nuclear Regulatory Commission Request for Additional Information Regarding License Amendment Request to Adopt 10 CFR 50.69

Waterford Steam Electric Station, Unit 3
NRC Docket No. 50-382
Renewed Facility Operating License No. NPF-38

References:

1) Entergy Operations, Inc. (Entergy) letter to U. S. Nuclear Regulatory Commission (NRC), "Application to Adopt 10 CFR 50.69, "Risk-Informed Categorization and Treatment of Structure, System and Components (SSCs) for Nuclear Power Reactors", W3F1-2020-0047, (ADAMS Accession No. ML20353A433), dated December 18, 2020

2) U. S. NRC to Entergy email, "Final RAIs to Entergy Operations, Waterford Steam Electric Station, Unit 3 – LAR to Adopt 10 CFR 50.69. (EPID L 2020 LLA-0279)," (ADAMS Accession No. ML21218A040) dated July 26, 2021

By letter dated December 18, 2020 (Reference 1), Entergy Operations Inc., (Entergy) requested an amendment to Appendix A, "Technical Specifications" (TS) of Renewed Facility Operating License NPF-38 for Waterford Steam Electric Station, Unit 3 (Waterford 3) to adopt 10 CFR 50.69.

By email correspondence dated July 26, 2021 (Reference 2), the NRC staff informed Entergy that they have reviewed the license amendment request and have determined that additional information is required to complete the review. A clarification call between the NRC and Entergy was previously held on July 7, 2021.

The additional information requested by the NRC in Reference 2 is provided in Enclosures 1 and 2 to this letter.

This letter contains new regulatory commitments listed in Enclosure 3.

Should you have any questions or require additional information, please contact Paul Wood, Waterford 3 Regulatory Assurance Manager, at 504-464-3786.

I declare under penalty of perjury; the foregoing is true and correct.
Executed on October 1, 2021.

Respectfully,

Ron Gaston

RWG/rrd

Enclosure 1: Responses to APLA RAIs

Enclosure 2: Responses to APLC RAIs

Attachment 1 to Enclosure 2, PSA-WF3-04-01

Enclosure 3: List of Regulatory Commitments

cc: NRC Region IV Regional Administrator
NRC Senior Resident Inspector – Waterford 3
NRC Project Manager Waterford 3
Louisiana Department of Environmental Quality, Office of Environmental Compliance

Enclosure 1 to

W3F1-2021-0050

Responses to APLA RAIs

Waterford 3 50.69 LAR

Responses to APLA RAIs

By letter dated December 18, 2020 (Agencywide Documents Access and Management System (ADAMS) Accession No. ML20353A433), Entergy Operations, Inc (Entergy or the licensee) submitted a license amendment request (LAR or the application) for the use of a risk-informed process for the categorization and treatment of structures, systems, and components at Waterford Steam Electric Station, Unit 3 (Waterford). The proposed license amendment would modify the Waterford licensing basis, by the addition of a License Condition, to allow for the implementation of the provisions of Title 10 of the Code of Federal Regulations (CFR), Part 50.69, "Risk-Informed Categorization and Treatment of Structures, Systems and Components for Nuclear Power Reactors." The U.S. Nuclear Regulatory Commission (NRC) staff from Probabilistic Risk Assessment Licensing Branch A (APLA) has reviewed the LAR and requests additional information (RAI) in order to complete the review.

APLA RAI 01 – Open Internal Events PRA Facts and Observations (F&O)

Section 50.69(c)(i) of 10 CFR requires that a licensee's PRA must be of sufficient quality and level of detail to support the categorization process and must be subjected to a peer review process assessed against a standard or set of acceptance criteria that is endorsed by the NRC. Section 50.69(b)(2)(iii) of 10 CFR requires that the results of the peer review process conducted to meet 10 CFR 50.69 (c)(1)(i) criteria be submitted as part of the application.

Regulatory Guide (RG) 1.200, Revision 2[1] provides guidance for addressing PRA acceptability. RG 1.200, Revision 2, describes a peer review process using the American Society of Mechanical Engineers/American Nuclear Society (ASME/ANS) PRA standard ASME/ANS-RA-Sa-2009[2], as one acceptable approach for determining the technical acceptability of the probabilistic risk assessment (PRA). The primary results of peer review are the Facts and Observations (F&Os) recorded by the peer review team and the subsequent resolution of these F&Os. A process to close finding-level F&Os is documented in Appendix X to the Nuclear Energy Institute (NEI) guidance documents NEI 05-04, NEI 07-12, and NEI 12-13,[3] which was accepted by the NRC.[4]

Section 1-A.2 of the 2009 PRA standard defines an PRA Upgrade as a method as new to the PRA model and Example 24 of the non-mandatory appendix states a new Human Reliability Analysis (HRA) approach would constitute an PRA Upgrade.

LAR Enclosure, Attachment 3 presents the dispositions for two F&Os that remain open after the F&O closure review (F&Os HR-F2-01 and HR-G4-01) which were assessed by the F&O closure review team as partially resolved based on the updates to the Human Reliability Analysis (HRA) spreadsheets. Both dispositions presented in the LAR state that the Waterford HRA was subsequently included the use of the EPRI HRA calculator to perform the human reliability analysis. The NRC staff notes that the HRA calculator has the following HRA methods and inputs: HCR, ORE, CBDTM, PSFs, and stress levels in addition to ASEP and THERP. It is

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1 Regulatory Guide 1.200, "An Approach for Determining the Technical Adequacy of Probabilistic Risk Assessment for Risk-Informed Activities," Revision 2, March 2009 (ADAMS Accession No. ML090410014).

2 American Society of Mechanical Engineers (ASME) and American Nuclear Society (ANS) PRA standard ASME/ANS RA-Sa-2009, "Addenda to ASME/ANS RA-S-2008, Standard for Level 1/Large Early Release Frequency Probabilistic Risk Assessment for Nuclear Power Plant Applications", February 2009, New York, NY (Copyright).

3 Anderson, V.K., Nuclear Energy Institute, letter to Stacey Rosenbergy, U.S. Nuclear Regulatory Commission, "Final Revision of Appendix X to NEI 05-04/07-12-12-16, Close-Out of Facts and Observations,", February 21, 2017 (ADAMS Accession No. ML17086A431).

4 Giitter, J., and Ross-Lee, M.J., U.S. Nuclear Regulatory Commission, letter to Krueger, G., Nuclear Energy Institute, "U.S. Nuclear Regulatory Commission Acceptance on Nuclear Energy Institute Appendix X to Guidance 05-04, 07-12, and 12-13, Close-Out of Facts and Observations (F&Os)", May 3, 2017 (ADAMS Accession No. ML17079A427).

unclear to the staff what HRA methods were used in both the spreadsheets and the HRA Calculator[5]. In light of these observations:

a. Describe the HRA methods used in the HRA spreadsheets and the HRA Calculator.

b. Provide justification that the implementation of the HRA Calculator in the Waterford PRA does not constitute a PRA Upgrade as defined in the ASME/ANS 2009 PRA standard. To support this justification, include discussion on whether the numerical differences between the peer-reviewed HRA methods and the EPRI HRA Calculator were compared during this HRA update and summarize the outcome of the differences.

c. Alternatively to Part (b), propose a mechanism to ensure a focused-scope peer review is conducted on the new HRA methods and all associated F&Os closed by the Appendix X approved process prior to implementing the 10 CFR 50.69 categorization process.

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[5] Table 6 of PSA-WF3-01-HR, Revision 3 appears to state the CBDTM/HCR Combination (Max) method was used.

<u>Waterford 3 Response</u>

a. The HRA methods used in the HRA Toolbox spreadsheets for post initiator actions are HCR/ORE and CBDTM for post initiator actions depending on the highest value. For pre-initiators, ASEP and THERP methodologies are applied. The HRA methods applied in the HRA calculator are HCR/ORE and CBDTM for post-initiators and ASEP/THERP for pre-initiators. In addition to these methods, the HRA toolbox spreadsheets employ performance shaping factors and stress levels. These items are also within the capability of the HRA Calculator.

b. The ASME/ANS 2009 PRA Standard defines a PRA Upgrade as the incorporation into the model of a new methodology or significant change in scope or capability of the PRA model that affects significant accident sequences or significant accident progression sequences. As seen in the response to RAI APLA 1a., no change in methods occurred from the transition to the HRA Calculator. From PSA-WF3-01-HR Section 1.2, the following statement is made regarding the transition to the HRA Calculator:

> *"A number of [human failure events] HFEs were updated, timing in particular, following updates in the MAAP thermo-hydraulic analysis (9). In addition, a different assessment tool was used -- the EPRI HRA Calculator software was employed instead of the Toolbox methodology used in the previous HRA. It should be noted that this conversion should not be considered a change in methodology, but rather a change of an assessment tool. Both the Toolbox and the HRA Calculator perform the assessments following methodology and guidance outlined in NUREG/CR-1278 (10), NUREG/CR-4772 (11), NUREG-1624 (12), NP-6560L (13), TR-100259 (14), and TR-101711 (6). Therefore, the methodology of the assessment remained unchanged and HRA Calculator is simply an improved assessment tool that offers a more structured approach."*

A comparison of the previous revision HFE with the post HRA Calculator transition HFE values was performed during the Revision 6 model update. The electronic file containing the comparison was saved/retained, but this comparison was not part of any formal document. This review showed that in some instances, the HRA Calculator would give a different factor for the same choice, however, these differences were almost exclusively in the conservative direction. In addition to the transition to the HRA Calculator, this model update included re-analysis of several HFEs to include updated timing and other inputs in order to resolve the F&Os listed in this question. This re-analysis led to the majority of changes in HFE values in the update.

To assess if the transition to the HRA Calculator had any impact on significant accident sequences, the top ten HFEs from Rev. 5 (PSA-WF3-01, R0) and the top ten HFEs from Rev. 6 (PSA-WF3-01, R2) were investigated to determine if any significant changes occurred solely due to the transition. (Table 1-1 and 1-2) As seen in the final column of these tables, the change in each HFE was due to changes or refinements to analysis inputs (and not due to the transition to HRA Calculator). The reason for these changes was due to updates to the inputs for timing and recoveries, not due to the transition to the HRA Calculator. Given that the significant HFEs did not change significantly due to the transition to HRA Calculator, no appreciable changes occurred to significant accident sequences and accident progression sequences.

Table 1-1: Revision 5 Top 10 Operator Actions							
R5 Event Name	**R6 Event Name**	**Description**	**R5 FV**	**R5 Prob**	**R6 Prob**	**% increase**	**Reason for Change**
DHFBAT_LSP	DC--XHE-FO-LDSHD	Shed battery loads for A or B or AB battery	1.05E-01	8.35E-02	3.03E-03	-96.37%	Updated timing and dependency credit for self review
QHFCSPWCTP	EFW-XHE-FO-EFW-WCT	Align EFW suction to WCT after CSP depletion	9.90E-02	3.85E-03	3.85E-04	-90.01%	Updated timing and dependency credit for self review
QHFCSPEMPP	EFW-XHE-FO-CSPMK	Makeup to CSP during EFW operation	9.75E-02	3.20E-05	1.00E-05	-68.77%	Changes to cognitive and execution recoveries
OHFRESTFWP	FW--XHE-FO-FW	Restore feedwater (e.g., via auxiliary feedwater)	5.35E-02	2.54E-03	1.64E-02	544.61%	Updated analysis based on procedure and timing updates
FHFCSTMAKP	CMU-XHE-FO-CSTMU	Makeup to the CST thru the condenser or DWST	4.96E-02	2.98E-05	1.00E-05	-66.49%	Changes to cognitive and execution recoveries
OHFCONDSTP	CD--XHE-FO-SGFEED	Restore feed to steam generators via CD pumps	3.86E-02	1.39E-03	1.41E-02	911.22%	Changes to dependency credit for other crew recovery
OHFRCPTRIP	RC--XHE-FO-RCP-SEAL	Trip RCPs following loss of seal cooling	2.46E-02	2.22E-03	1.56E-03	-29.71%	Changes to cognitive and execution recoveries
HHFISOMINP	SI--XHE-FO-HPSI-REC	Isolate HPSI pump recirc lines after RAS	1.74E-02	1.00E-05	1.00E-05	0.00%	No change
QHFEFWFLOP	EFW-XHE-FO-MAN	Control EFW Manually (FCV Valves)	1.51E-02	1.07E-02	4.27E-03	-60.09%	Changes to cognitive and execution recoveries
QHFEFWLOOP	EFW-XHE-FO-LOOP	Establish EFW Flow after power restoration	1.48E-02	8.20E-04	1.48E-02	1704.88%	Changes to dependency credit for execution recoveries

Table 1-2: Revision 6 Top 10 Operator Actions[1]

R6 Event	R5 Event	Description	R6 FV	R6 Prob	R5 Prob	% increase	Reason for Change
RC--XHE-FO-RCP-SEAL	OHFRCPTRIP	Trip RCPs following loss of seal cooling	1.41E-01	1.56E-03	2.22E-03	-29.71%	Changes to cognitive and execution recoveries
CC--XHE-FO-AB-TRNST	SHFABCCWTP	Align CCW train AB to replace lost train A or B (Trans)	8.03E-02	5.10E-04	1.90E-04	168.42%	Changes to dependency credit for other crew recovery
CC--XHE-FO-TRNISOL	SHFTRNISOP	Isolate trains during single CCW pump operation	7.37E-02	1.00E-05	2.83E-05	-64.68%	Changes to cognitive and execution recoveries
EFW-XHE-FO-EFW-WCT	QHFCSPWCTP	Align EFW suction to WCT after CSP depletion	4.79E-02	3.85E-04	3.85E-03	-90.01%	Updated timing and dependency credit for self review
FW--XHE-FO-FW	OHFRESTFWP	Restore feedwater (e.g., via auxiliary feedwater)	4.26E-02	1.64E-02	2.54E-03	544.61%	Updated analysis based on procedure and timing updates
EFW-XHE-FO-CSPMK	QHFCSPEMPP	Align makeup to CSP during EFW operation	4.22E-02	1.00E-05	3.20E-05	-68.77%	Changes to cognitive and execution recoveries
SI--XHE-FO-HPSI-REC	HHFISOMINP	Isolate HPSI pump recirc lines after RAS	2.22E-02	1.00E-05	1.00E-05	0.00%	No change
EG--XHE-FO-TEDG	EHF-TEDG-P	Operator fails to start/ align/load TEDG	1.61E-02	9.57E-02	2.68E-03	3475.98%	Updated analysis based on procedure and timing updates
4KV-XHE-FO-SUT	EHFMANTRNP	Transfer loads to Startup Xfmr w/ auto transfer fails	1.61E-02	2.62E-01	1.43E-02	1732.30%	Updated detailed analysis performed for internal and fire
4KV-XHE-FO-AB3S	EHFALNAB_P	Energize bus 3AB3-S opposite supply (Trans)	1.50E-02	1.60E-02	2.33E-03	587.73%	Increase due to changes made to system time window

Note 1: This list displays the top 10 post-initiator HFEs. One pre-initiator was identified that made the top ten: SI--XHE-MC-LT0305, miscalibration of RWSP level transmitters. Its value did increase during the model update; however, this was a result of changing the assumed population of transmitters tested each time.

c. As discussed in the Waterford response to RAI APLA 1b, the transition from the HRA Toolbox spreadsheets to the HRA Calculator does not constitute a PRA Upgrade as defined in the ASME/ANS 2009 PRA standard. Entergy does not intend to conduct a focused-scope peer review on the updated HRA methods and all associated F&Os closed by the Appendix X approved process prior to implementing the 10 CFR 50.69 categorization process.

APLA RAI 02 - Process for Review of Key Assumptions and Sources of Uncertainty in the internal events PRA

RG 1.174, Revision 3[6] describes an approach that is acceptable to the NRC staff for developing risk-informed applications for a licensing basis change that considers engineering issues and applies risk insights. It provides general guidance concerning analysis of the risk associated with the proposed changes in plant design and operation. Section C.6 of RG 1.200 provides guidance regarding documentation of the acceptability of the PRA to support a regulatory submittal. Further, Section 2.5 of RG 1.174 states that the impact of PRA uncertainties should be considered, including uncertainties that are explicitly accounted for in the results and those that are not, and cites NUREG-18556[7] provides acceptable guidance for the treatment of uncertainties in risk-informed decision-making.

NUREG-1855 describes how the impact of PRA uncertainties should be assessed and documented. It states, "Additional qualitative screening criteria may be identified as applicable for specific applications. The bases for any criteria used to qualitatively eliminate missing scope and level-of-detail items from a PRA must be documented", as well as, "At a minimum, assumptions made in lieu of data, operational experience or design detail should be well documented with the basis for the assumptions clearly explained."

LAR Attachment 6 describes the process used for reviewing the PRA assumptions and sources of uncertainty. The staff reviewed the Waterford uncertainty documents during the regulatory audit[8] for the internal events, internal flooding and fire PRA. With regards to the internal events PRA, address the following:

a. Describe the process used for reviewing the PRA key assumptions and sources of uncertainty for the application for the internal events PRA.

b. Explain whether and how the plant specific PRA assumptions and sources of uncertainty were assessed during this review for impact on the 50.69 application.

c. Confirm that the review of plant specific PRA assumptions and sources of uncertainties was documented for use in the 50.69 categorization program, or alternatively, propose a mechanism to ensure that this review is documented prior to the implementation of the 10 CFR 50.69 categorization.

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6 Regulatory Guide 1.174, "An Approach for Using Probabilistic Risk Assessment in Risk-Informed Decisions on Plant Specific Changes to the Licensing Basis," Revision 3, January 2018 (ADAMS Accession No. ML17317A256).

7 NUREG-1855, “Guidance on the Treatment of Uncertainties Associated with PRAs in Risk-Informed Decisionmaking”, Revision 1

8 Buckberg, P., U.S. Nuclear Regulatory Commission, letter to Site Vice President, Entergy Operations, Inc. - Waterford Steam Electric Station, Unit 3, "WATERFORD STEAM ELECTRIC STATION, UNIT 3 – REGULATORY AUDIT IN SUPPORT OF REVIEW OF APPLICATION TO ADOPT 10 CFR 50.69, “RISK-INFORMED CATEGORIZATION AND TREATMENT OF STRUCTURES, SYSTEM, AND COMPONENTS FOR NUCLEAR POWER REACTORS”" (EPID L-2020-LLA-0279) (ADAMS Accession No. ML21099A002)

Waterford 3 Response

a. The process is described in the LAR Attachment 6. A specific report was prepared to assess PRA quality and the key assumptions and sources of uncertainty from the PRA analyses for internal events, internal floods, and internal fires. This report (PSA-WF3-08-06) highlighted the key sources of uncertainty from the various applications and determined if they are key to the 50.69 application. This report provided by inclusion the specific key assumptions and sources of uncertainty from the internal events (and other hazards) uncertainty analysis that warrant further consideration for applications. The internal events sensitivity and uncertainty analysis, PSA-WF3-01-QU-01, is the input for consideration of key assumptions and sources of uncertainty that should be screened in for applications.

   With respect to the internal events uncertainty analysis, PSA-WF3-01-QU-01, the primary source of potential key assumptions (and sources of uncertainty) was Table A3-1 listing generic sources of uncertainty from the referenced EPRI report and their consideration in the model. Note, while the EPRI list is generic in nature, the Waterford assessment of these is site specific. Additionally, topics for which sensitivity studies were performed were also reviewed for inclusion as potential key assumptions/sources of uncertainty to the application. The sensitivity studies in Section 3.1 of this report are based on a review of major model assumptions from the WF3 Internal Events System Notebook package as well as recent changes made to the model. Additionally, pending model changes are also considered for sensitivity studies.

b. A specific review of plant specific assumptions and sources of uncertainty was not explicitly performed as it was assumed to be inherent in the review of the sensitivity study items as they were the final items determined to be "key". Of the key assumptions and sources of uncertainty retained for disposition in the LAR, three are plant specific items derived from the sensitivity studies in PSA-WF3-01-QU-01 (credit for FLEX, credit for TEDG, and variable cooling tower alignments). While the current approach likely captures all of the relevant items, for completeness, Entergy will ensure that prior to categorization, PSA-WF3-08-06 is updated to include a review of plant specific assumptions to determine if any would be considered "key" for the application per the NUREG 1855 Rev. 1 guidance.

   The key assumptions and sources of uncertainty discussed in the LAR were determined based on a screening of the key assumptions and sources of uncertainty in the internal events uncertainty analysis. Each key assumption was evaluated to determine if it had the potential to be key for the application. Per the guidance in NUREG 1855, key assumptions involving consensus methods were screened out and did not warrant further consideration. Additionally, key assumptions that had an obviously negligible impact to the results were screened as well. These items were based on engineering judgement and the criteria were not explicitly documented. As described in the answer to RAI APLA 01a., plant specific assumptions were not explicitly reviewed, rather the review of sensitivity studies was assumed to include the key plant specific assumptions. Entergy will ensure that prior to categorization, PSA-WF3-08-06 is updated to include listing of specific screening criteria for screening key assumptions as potential key assumptions for the application. This update will include the documentation of the process utilized from NUREG 1855 Revision 1.

c. The systematic review of key assumptions and sources of uncertainty in PSA-WF3-08-06 will be updated to include more thorough documentation of the process used prior to categorization of any SSCs. This update will include a review of plant specific assumptions to verify that all key assumptions and sources of uncertainty are captured for disposition for consideration in this application. The screening process documentation will be updated to list the specific screening criteria used of determining if assumptions and sources of uncertainty that are "key" for their respective hazards are also potentially "key" for the application.

Specific criteria to be used for assumptions and uncertainties to be screened from further consideration:

- The uncertainty or assumption will have no impact on the PRA results and therefore no impact on the decision of HSS or LSS for any SSCs.
- There is no different reasonable alternative to the assumption which would produce different results and/or there is no reasonable alternative that is at least as sound as the assumption being challenged. (RG1.200 Rev 2)
- The uncertainty or assumption implements a conservative bias in the PRA model, and that conservatism does not influence the results. These conservatisms are expected to be slight and only applied to minor contributors to the overall model. EPRI 1013491 uses the term "realistic conservatisms." Thus, uncertainties/assumptions that implement realistic [slight] conservativisms can be screened from further consideration.
- EPRI 1013491 elaborates on the definition of a consensus model to include those areas of the PRA where extensive historical precedence is available to establish a model that has been accepted and yields PRA results that are considered reasonable and realistic. Thus, uncertainties/assumptions where there is extensive historical precedence that produces reasonable and realistic results can be screened from further consideration.

If the assumption or uncertainty does not meet one of the criteria above, then it is retained as "key" for the application.

APLA RAI 03 - Dispositions of Key Sources of Uncertainty

Paragraphs (c)(1)(i) and (ii) of 10 CFR 50.69 require that a licensee's PRA be of sufficient quality and level of detail to support the SSC categorization process, and that all aspects of the integrated, systematic process used to characterize SSC importance must reasonably reflect the current plant configuration and operating practices, and applicable plant and industry operational experience. The guidance in NEI 00-04 specifies that sensitivity studies to be conducted for each PRA model to address uncertainty. The sensitivity studies are performed to ensure that assumptions and sources of uncertainty (e.g., human error, common cause failure, and maintenance probabilities) do not mask importance of components. NEI 00-04 guidance states that additional "applicable sensitivity studies" from characterization of PRA adequacy should be considered.

The dispositions provided in LAR Attachment 6 for some of the key assumptions or sources of uncertainty appear to potentially impact the SSC categorization process.

a) Item # 3 in LAR Table 6-2 identifies fire frequencies for ignition sources as a Fire PRA (FPRA) source of uncertainty since the Waterford FPRA utilizes the frequencies from the EPRI Supplement 1 to NUREG-6850 and credit for detection and suppression. The sensitivity study documented in the fire PRA uncertainty document audited by the NRC staff [9] appears to show significant increases in core damage frequency (CDF) and large early release frequency (LERF) risks when the original NUREG-6850 values were used. However, the NRC staff notes that updated fire ignition frequencies have been published. [10]

   i. Provide a detailed justification for why the ignition frequencies "will not have an appreciable impact on the 10 CFR 50.69 categorization". Provide technical justification for its use and evaluate the significance of its use on the risk metrics for the application (RG 1.174) provided in Attachment 2 of the LAR.

   ii. Alternatively to part (i), propose a mechanism to incorporate the updated fire ignition frequencies in the fire PRA model prior to the implementation of the 10 CFR 50.69 categorization process.

b) Item # 2 in LAR Table 6-2 identifies exclusion of certain systems due to lack of cable data a FPRA source of uncertainty. The LAR further states that "the current approach used (assume equipment lacking detailed cable data is failed) will result in conservative evaluations" and "this conservatism would tend to result in additional SSCs being categorized as High Safety Significant in the 10 CFR 50.69 categorization process."

   Describe the type of systems assumed failed in the FPRA and provide further justification as to why this assumption will not impact 10 CFR 50.69 categorizations.

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[9] Case 6 from the Waterford PSA-WF3-UNC-01, Revision 0 Notebook.

[10] NUREG-2169, "Nuclear Power Plant Fire Ignition Frequency and Non-Suppression Probability Estimation Using the Updated Fire Events Database: United States Fire Event Experience Through 2009."

Waterford 3 Response

a.) i. Waterford completed a sensitivity analysis to examine the impact of updating the Fire PRA ignition frequencies to NUREG-2169 values. The sensitivity case examined both the impact the change would have on overall risk results, and the impact the change would have on 50.69 component categorization.

Ignition frequencies for all fire scenarios were updated to NUREG-2169 values. There were no other changes made to the individual fire scenarios, such as oil/electrical split fractions. Fire targets, heat release rates, fire detection and/or suppression credit, and area weighting factors were all identical in the two cases evaluated (the baseline case and the sensitivity case).

The following table shows the results of the sensitivity case:

Case	CDF	LERF
Baseline Fire PRA (Supp 1)	2.02E-05	2.06E-07
IGF based on NUREG-2169	5.61E-05	6.08E-07

The Fire PRA CDF and LERF results are greatly impacted by the ignition frequencies applied. With the updated frequencies, the CDF increased by a factor of 2.5 and LERF increased by nearly a factor of 3. The choice of ignition frequencies has a very significant impact on overall risk numbers. Note that the results would be improved by utilizing the new HRRs in NUREG-2178, which is generally implemented at the same time as NUREG-2169.

Importance Measures were also evaluated to examine the impact the ignition frequency sensitivity would have on categorization results.

All Fussell-Vesely (FV) and Risk Achievement Worth (RAW) importance measures were checked in the base fire case and in the ignition frequency sensitivity case. NEI 00-04 guidance provides the criteria to use on these PRA importance measures to identify component safety significance. From NEI 00-04:

The importance measure criteria used to identify candidate safety significance are:

- *Sum of F-V for all basic events modeling the SSC of interest, including common cause events > 0.005*
- *Maximum of component basic event RAW values > 2*
- *Maximum of applicable common cause basic events RAW values > 20*

If any of these criteria are exceeded, it is considered candidate safety-significant.

The population of basic event (including common cause event) RAW values that exceed 2.0 (20.0 for common cause events) was compared for the two sets of results. The comparison was made for both CDF and LERF results. Events that had RAW values exceeding the criteria in one case but not the other were identified. This was done for both – high RAW values in the baseline case but not in the sensitivity case, and high RAW values in the sensitivity case but not in the baseline case. This comparison shows the difference in potential categorization results from RAW values based on the change to ignition frequencies.

A similar effort was completed for FV importance measures. FV is different for 50.69 evaluation since FV values for multiple events related to a single component are considered together. The FV importance of a component is the sum of the FV importances for all the failure modes of the component relevant to the function being evaluated. So, an additional step was included in the FV comparison to map each basic event to a component and ensure FV values for components with multiple basic events were summed. Once the FV values were mapped and summed, the same comparison was made for both CDF and LERF results to see which components exceeded the NEI 00-04 FV criteria in one case but not the other.

The results of the importance measure comparison showed the impact the ignition frequency update would have on 50.69 categorization. The resulting lists of component and basic events showed that there is essentially no impact to categorization results based on changing only the ignition frequencies.

There are no differences in the population of high safety significant components when only FV measures are compared. All components with FV values more than 0.005 in the IGF sensitivity case are also high safety due to FV in the baseline case. There is a single component that has a FV more than 0.005 in the baseline case that has a FV less than 0.005 in the IGF sensitivity case, but that component was high safety significant due to sensitivity case RAW.

The classification of two power-related (DC) and two EFW system basic events that have RAW values more than 2 in the baseline case have RAW values less than 2 in the IGF sensitivity case. However, the systems to which these basic events are mapped are systems that would not be reasonable candidates for 50.69 categorization as they are HSS in the internal events model.

The classification of several power-related (SSD, 4KV, and ID) basic events and CCF events that do not exceed the RAW thresholds in the baseline case exceed the RAW values more than 2 in the IGF sensitivity case. However, the systems to which these basic events are mapped are systems that would not be reasonable candidates for 50.69 categorization as they are HSS in the internal events model. For all but one of these basic events, the full power internal events (FPIE) baseline RAW values are more than 2, so they are not candidates for 50.69 categorization. For the remaining basic event, other basic events in the system are determined to be of high safety significance in the FPIE baseline case, and therefore this system is not a candidate for 50.69 categorization.

Based on the comparison of CDF and LERF importance measures for the IGF sensitivity, the source of ignition frequencies for the Fire PRA has a significant impact on total risk. CDF and LERF values are both considerably higher with NUREG-2169 fire frequencies applied to the PRA model. The impact that the ignition frequency data source choice has on 50.69 categorization is much less significant, and in fact, there is almost no impact to a 50.69 categorization results based on the treatment of IGF components. The existing IGF sensitivity case, which models the NUREG-2169 ignition frequencies, results in more components considered to be of high safety significance in systems that would not be reasonable candidates for 50.69 categorization, such as 4KV and SSD. Therefore, using ignition frequencies from an older source (i.e., NUREG/CR-6850) in the FPRA model provides negligible 50.69 related importance measure masking. Also, all the additional components determined to of high safety significance in the IGF sensitivity case are already determined to be such due to the FPIE PRA 50.69 categorization or are part of a system that is categorized as high.

The sensitivity completed for this response shows that the use of dated ignition frequencies for the FPRA does not have an appreciable impact on categorization results. The next Fire PRA model revision will include an update to ignition frequencies. However, categorization results are not impacted based on an assessment of the most recent ignition frequencies, and categorization efforts competed prior to the next Fire PRA update are not impacted by this fire model limitation. If any of the systems for which this sensitivity case shows any changes in the safety significance for the fire PRA model (EFW, DC, ID, SSD, 4KV) are selected for categorization prior to the update of the ignition frequencies to the industry consensus approach (currently NUREG 2169), the results will be shared with IDP members during review to ensure they are both aware of the model limitations, but also that these limitations are related to a limited subset of components.

The Fire PRA results do show a notable increase in both CDF and LERF when updated ignition frequencies are applied. However, the increased CDF and LERF values do not change any conclusions or invalidate anything contained in the LAR. It shows a higher site risk associated with the internal fire hazard but does not have an appreciable impact on the assessment or significance regarding the risk metrics for the application (RG 1.174).

a) ii For categorization efforts related to the impacted systems (EFW, DC, ID, SSD, 4KV) completed prior to the next Fire PRA update (and application of updated ignition frequencies), the results of this sensitivity case will be shared with IDP members during review to ensure they are both aware of the FPRA model limitations, but also aware that it does not have an appreciable impact on results.

b). Waterford completed a sensitivity analysis to examine the impact of failing components with ‘Unknown Location’ (UNL) - insufficient or incomplete cable location data. The systems with components included in the UNL list (failed in all fire scenarios) include Main Steam, Main Feedwater, Condensate, Instrument Air, and Station Air. This modeling treatment treats all components with unknown location as failed in all fire scenarios. This is a conservative treatment. The sensitivity was completed to check to see what impact this treatment has on importance measures and 10 CFR 50.69 categorization.

The Fire PRA model was quantified with nominal treatment (UNL components failed in every scenario). A sensitivity case was then completed with the UNL feature not applied. In this case, all the UNL related components are not impacted by fire and only subject to random failure.

The following table shows the results of the sensitivity case:

Case	CDF	LERF
Baseline Fire PRA (UNL components failed)	2.02E-05	2.06E-07
UNL sensitivity (UNL components not failed)	1.92E-05	1.98E-07

Based on the results, failing components included in UNL has a small conservative impact on Fire PRA CDF and LERF. Not failing UNL components in the model results in a decrease in CDF of approximately 5% and a decrease in LERF of approximately 4%. Importance Measures were also evaluated to examine the impact the UNL sensitivity would have on categorization results.

All Fussell-Vesely (FV) and Risk Achievement Worth (RAW) importance measures were checked in the base fire case and in the UNL sensitivity case. NEI 00-04 guidance provides the criteria to use on these PRA importance measures to identify component safety significance. From NEI 00-04:

The importance measure criteria used to identify candidate safety significance are:

- *Sum of F-V for all basic events modeling the SSC of interest, including common cause events > 0.005*
- *Maximum of component basic event RAW values > 2*
- *Maximum of applicable common cause basic events RAW values > 20*

If any of these criteria are exceeded, it is considered candidate safety-significant.

The population of basic event (including common cause event) RAW values that exceed 2.0 (20.0 for common cause events) was compared for the two sets of results. The comparison was made for both CDF and LERF results. Events that had RAW values exceeding the criteria in one case but not the other were identified. This was done for both – high RAW in the baseline case but not in the sensitivity case, and it was done to identify high RAW values in the sensitivity case but not in the baseline case. This comparison shows the difference in potential categorization results from RAW values based on the change to credited UNL equipment.

A similar effort was completed for FV importance measures. FV is different for 50.69 evaluation since FV values for multiple events related to a single component are considered together. The FV importance of a component is the sum of the FV importance's for all the failure modes of the component relevant to the function being evaluated. So, an additional step was included in the FV comparison to map each basic event to a component and ensure FV values for components with multiple basic events were summed. Once the FV values were mapped and summed, the same comparison was made for both CDF and LERF results to see which components exceeded the NEI 00-04 FV criteria in one case but not the other.

The results of the importance measure comparison showed the impact the UNL treatment would have on categorization. The resulting lists of component and basic events showed that there no appreciable impact to categorization results based on changing only the UNL treatment for FPRA.

There are events/components that exceed the importance measure 10 CFR 50.69 criteria in the base model, but do not exceed the criteria in the sensitivity case (UNL turned off). There are however, no components that exceed the importance measure criteria in the sensitivity case that do not also exceed the criteria in the base Fire PRA case (this is true for both CDF and LERF comparisons). This shows that the UNL treatment is conservative and represents the treatment that includes the most limiting categorization results (most components classified as candidate safety significant).

Additionally, comparing the components/events that have potential different 10 CFR 50.69 categorization results based on UNL treatment to those importance measures for the same components in the at power internal events model – shows that almost all the components impacted by the sensitivity would already result in safety significant classification based on the base PRA model importance measures. Based on the sensitivity and assessment of importance measures, the UNL treatment in the Fire PRA has no meaningful impact on 10 CFR 50.69 categorization and the impact it does have has a conservative bias.

There are no differences in the population of components that exceed the FV values when the measures are compared. All components with a FV > 0.005 FV in the UNL sensitivity case also exceed the threshold in the baseline case. There is a single component with a high FV in the baseline case that is not high due to FV in the UNL sensitivity case, but that component also has a basic event that exceeds the RAW threshold (RAW > 2.0).

There is a single difference in the population CCF events (events with RAW > 20) between the baseline case and the UNL sensitivity case, and that one event has a 19.97 RAW in the baseline case and a 20.88 RAW in the UNL sensitivity case. It does represent an event that did pass the limit in the UNL sensitivity case, but with a 19.97 RAW it would have been considered for safety significant classification with a value that close to the threshold.

All (non-CCF) basic events with a high RAW (> 2.0) in the UNL sensitivity case also have high RAW in the baseline case. In other words, the baseline case results in the maximum number of 50.69 components being classified ‘safety significant’ and there is no impact based on the UNL treatment. However, there are several basic events that have high RAW in the baseline case that do not have high RAW values in the UNL sensitivity case. The systems to which these basic events are mapped (4KV, EFW, ID, ST, and UAT) are systems that would not be reasonable candidates for 50.69 categorization due to their safety significance in the internal events model. Also, for all but one of these basic events, the FPIE baseline RAW values are above the 2.0 RAW threshold, so they already meet safety significant criteria (so there is not impact from Fire PRA classification). There is a single basic event that both has notable RAW value differences in the FPRA baseline and sensitivity cases and is not already a candidate safety significant component based on the FPIE RAW results (a SUPS inverter that has RAW of more than 2 in the FPRA baseline case and less than 2 in the sensitivity case, and also has a RAW of less than 2 based on the FPIE PRA model).

Based on the comparison of CDF and LERF importance measures for the UNL sensitivity, there is no appreciable impact to a 50.69 categorization results based on the treatment of UNL components. The existing baseline case, which applies the UNL treatment, results in more safety significant components when importance measures are assessed. Also, the population of components are in systems that would not be reasonable candidates for 50.69 categorization, such as 4KV and EFW. Therefore, failing all UNL components in the FPRA model provides no 50.69 related importance measure masking and has a conservative bias (more components with RAW >2.0 in the case with UNL applied than in the case without UNL applied).

The results of the UNL sensitivity study do not change any conclusions or invalidate anything contained in the LAR and does not change any assessment or significance regarding the risk metrics for the application (RG 1.174).

The systems with components included in the UNL list include Main Steam, Main Feedwater, Condensate, Instrument Air, and Station Air. These systems are properly included in the FPIE model to accurately model the as-built as-operated plant. However, with fire scenarios often impacting normal power supply, several of these systems have very limited impact to the site response to fire events. The sensitivity case documented for UNL shows the exclusion of these components/events/systems from Fire PRA scenarios has very little impact on risk results or risk importance measures.

APLA RAI 04 – Crediting of FLEX in the PRA Model

The NRC memorandum dated May 30, 2017, "Assessment of the Nuclear Energy Institute 16-06, 'Crediting Mitigating Strategies in Risk-Informed Decision Making,' Guidance for Risk-Informed Changes to Plants Licensing Basis" (ADAMS Accession No. ML17031A269), provides the NRC's staff assessment of identified challenges and strategies for incorporating FLEX equipment into a PRA model in support of risk-informed decision-making in accordance with the guidance of RG 1.200. In the May 30, 2017 memo regarding equipment failure probability, the NRC staff concludes (Conclusion 8):

> The uncertainty associated with failure rates of portable equipment should be considered in the PRA models consistent with the ASME/ANS PRA Standard as endorsed by RG 1.200. Risk-informed applications should address whether and how these uncertainties are evaluated.

With regards to HRA, NEI 16-06 Section 7.5 recognizes that the current HRA methods do not translate directly to human actions required for implementing mitigating strategies. Sections 7.5.4 and 7.5.5 of NEI 16-06 describe such actions to which the current HRA methods cannot be directly applied, such as: debris removal, transportation of portable equipment, installation of equipment at a staging location, routing of cables and hoses; and those complex actions that require many steps over an extended period, multiple personnel and locations, evolving command and control, and extended time delays. In the May 30, 2017 memo, the NRC staff concludes (Conclusion 11):

> Until gaps in the human reliability analysis methodologies are addressed by improved industry guidance, [Human Error Probabilities] HEPs associated with actions for which the existing approaches are not explicitly applicable, such as actions described in Sections 7.5.4 and 7.5.5 of NEI 16-06, along with assumptions and assessments, should be submitted to NRC for review.

LAR Attachment 6 summarizes the credit for Diverse and Flexible Mitigation Capability (FLEX)S in the PRA. It states that the credit for FLEX equipment is limited to specific extended loss of offsite power scenarios. It further explains that only permanently installed FLEX equipment is credited, which includes a FLEX diesel generator to provide power to battery chargers and a FLEX Core Cooling Pump to provide Feedwater to the Steam Generators.

Item #1 of Table 6-1 in Attachment 6 of the LAR Enclosure identifies the incorporation of FLEX strategies and equipment in the PRA model as a source of uncertainty and has a sensitivity evaluation that demonstrates crediting FLEX in the model resulted in an impact on station blackout (SBO) risk. The results of the study[11] also demonstrate that the FLEX credit decreases CDF by seven percent. The disposition states that the inclusion of FLEX is not a source of uncertainty since it reflects the as-built, as-operated plant. The NRC staff notes the concern is in regard to the failure probabilities for FLEX equipment and operator actions. During the audit the NRC staff determined that safety-related failure data was used for the FLEX diesel generator failure rates. The NRC staff notes that industry generic data differentiates between safety and non-safety diesel generator failure rates due to their different pedigrees. The NRC staff also notes that the procedural cue for Extended Loss of AC Power (ELAP) declaration appears vague,

[11] Sensitivity Case #1 in the Waterford PSA-WF3-01-QU-01, Revision 2.

and that the ELAP declaration does not appear to be modeled. The NRC staff requests the following information:

a) LAR Attachment 3 states that generic failure data was judged applicable to the FLEX equipment because it is permanently installed and procedurally controlled. Justify the rationale for applying generic failure data to the FLEX equipment, and how the uncertainties associated with the parameter values are considered in the 50.69 categorization.

b) Describe the credited operator actions related to FLEX equipment and discuss the methodology used to assess the associated HEPs and the licensee personnel that performs these actions. The discussion should include:

   i. A summary of how the licensee evaluated the impact of the plant-specific human error probabilities and associated scenario-specific performance shaping factors listed in (a)-(j) of supporting requirement HR-G3 of ASME/ANS RA-Sa-2009, as endorsed by RG 1.200.

   ii. Regarding FLEX pre-initiators evaluation, address the following:

      (a) Whether maintenance procedures for the portable equipment were reviewed for possible pre-initiator human failures that renders the equipment unavailable during an event, and whether the probabilities of the pre-initiator human failure events were assessed as described in HLR-HR-D of ASME/ANS RA-Sa-2009, as endorsed by RG 1.200.

      (b) Alternatively to part (a) of this section, propose a mechanism to ensure incorporation of pre-initiator human failures in the PRA model prior to implementation of the 10 CFR 50.69 categorization.

   iii. Regarding FLEX strategy initiations, address the following:

      (a) Provide a discussion detailing the technical bases for probability of failure to initiate mitigating strategies. Include in this discussion the cue to enter ELAP and how it is incorporated into the PRA model used for categorization.

      (b) Alternatively to Part (a) of this section, propose a mechanism to ensure that entry into FLEX strategies is appropriately addressed and incorporated into the PRA model prior to the implementation of the 10 CFR 50.69 categorization.

c) Based on the Waterford PRA documentation audited[12] by the NRC staff, it appears that the four FLEX operator actions were removed from the HRA dependency analysis due to time differences. However, the NRC staff notes that the HRA Calculator Dependency Decision Tree tool designates Low Dependency for Moderate/High Stress levels independent of time or crew.

   i. Provide further discussion/justification for excluding the FLEX operator actions from the HRA dependency analysis.

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[12] Section 5.2 of Waterford PSA-WF3-01-HR, Revision 3.

ii. Clarify whether the WF3 HRA Dependency Analysis process was performed utilizing the HRA Calculator tools including the Dependency Decision Tree.

iii. Alternative to part i and ii of this question, propose a mechanism to include the FLEX actions in the PRA HRA

Waterford 3 Response

a) Generic data was judged as the best option for modeling FLEX equipment failures when the FLEX equipment and actions were added to the PRA model. Modeling of FLEX equipment and strategies includes only a few components. The list of components included in the FLEX logic includes:

FLEX Diesel Generator
FLEX Core Cooling Pump
FLEX Diesel Driven Fuel Transfer Pump
Transfer Switches
Disconnect Switches
Circuit Breakers
FLEX Room Exhaust Fan
FLEX Exhaust Damper
Battery Room Exhaust Fans

While the LAR stated that all credited equipment is permanently installed, it should be noted that the Diesel Driven Fuel Transfer Pump is portable. All other components represent permanently installed equipment. The circuit breakers, transfer switches, disconnect switches, exhaust fans, and damper are not notably different from any other similar/identical components modeled in the Waterford PRA model. The data and type codes used for these events is also applied to several other non-FLEX events throughout the PRA model. Based on the design, maintenance, and operation of these components, application of this data was judged reasonable.

The best option for these components was judged to be generic failure data. It was evident that this was a limitation and source of uncertainty, but Waterford judged it as the best option available. This model/data limitation is also limited to only three components. The equipment is maintained in the RAB, controlled by Waterford procedures, testing, and corrective action programs.

Waterford plans to apply the FLEX failure data from the PWROG-18042-P Revision 1 guidance currently being developed for consideration during the next model update.

Application of generic data for FLEX components will not have an appreciable impact on the 10 CFR 50.69 program. A sensitivity case was completed to examine the potential impact FLEX modelling has on 10 CFR 50.69 categorization. The case evaluated the base FPIE model with credit for FLEX compared to the results with no credit for FLEX. This shows the maximum impact FLEX modeling has on results and importance measures. The results of the sensitivity show that the treatment of FLEX has very little impact on overall PRA results (no credit at all for FLEX results is approximately a 5.5% increase in CDF and 2% increase in LERF). Additionally, credit for FLEX is limited to the Full Power Internal Events PRA model (FLEX is not credited in the Fire PRA model or the Internal Flooding PRA model).

The sensitivity case also included an assessment of the importance measures used for 10 CFR 50.69 categorization. The importance measures for the baseline case (with credit for FLEX) and the sensitivity case (no FLEX credit) were compared for CDF and LERF cutsets. The goal of the comparison was to examine if the population of events &

components that would be required to be candidate safety significant for 10 CFR 50.69 would be impacted by FLEX treatment.

Based on the NEI 00-04 categorization criteria for PRA components and events, the FLEX sensitivity has the following impact.

- Based on comparison of Fussell-Vesely (FV) importance measures between the baseline case and the FLEX sensitivity case, no classifications of components would be changed due to FLEX treatment based only on the FV results. A few components (TEDG, EDG Cooling Fans) did have FV differences that would put them in a different class. However, for each component, the corresponding RAW values for the basic events were above 2.0, so the safety significant classification would result regardless of the FV value.
- The classification of four relays that would be safety significant (RAB >2.0) in the baseline case would be potential non-safety significant classification - in the FLEX sensitivity case. However, all four listed components have a RAW of 1.95 so they would get some consideration for safety significant classification given the proximity to the 2.0 threshold (note – the four relays are ESFAS and EDG Relays and those systems are not likely systems for 10 CFR 50.69 categorization).
- The classification of basic events due to CCF events that are considered potentially not-safety significant in the baseline case and safety significant in the FLEX sensitivity case only occurred for components that were already HSS due to other metrics.

The comparison resulted in 20 components with importance measures above the NEI thresholds for the sensitivity case that were below the NEI thresholds in the base case. However, the 20 components are included in EFW or EDG systems or are in systems supporting EFW or EDG functions (example: HVR system exhaust fan for the EDG room cooling).

Based on the comparison of CDF and LERF importance measures for the FLEX sensitivity, the FLEX treatment has no impact on a 10 CFR 50.69 program. The only changes to importance measures that would impact categorization are all tied to systems that would not be reasonable candidates for 10 CFR 50.69 categorization. While the impact to categorization is minimal and related to systems that are not expected to be categorized, credit for FLEX equipment will be classified as a "key" source of uncertainty for EFW, EDG, and FLEX systems for this application. The disposition in Table 6-1 will be updated to reflect this.

Given that this is a key source of uncertainty relative to these systems, the results of this sensitivity study will be shared with the 10 CFR 50.69 Integrated Decision-making Panel (IDP) for categorization of these systems to ensure they are aware of the impact of FLEX modeling when making decisions. This will also ensure the IDP is aware of potential equipment/component impacts of FLEX treatment.

b) i The impact of the plant-specific human error probabilities and associated scenario-specific performance shaping factors listed in (a)-(j) of supporting requirement HR-G3 of ASME/ANS RA-Sa-2009 – were evaluated for the FLEX HEPs consistent with the methodology for development of all Waterford 3 PRA HEPs. This effort and a mapping of how the methodology meets the elements of HR-G3 is documented in PSA-WF3-01-HR.

There are no open peer review F&Os on this element and Waterford was judged to meet Capability Category II/III for supporting requirement HR-G3.

b) ii a) Maintenance procedures were not reviewed for FLEX equipment and no pre-initiator events are included in the FLEX modeling. This has been identified as a model limitation. A Model Change Request (MCR) has been initiated to track this technical issue and ensure it is captured (pre-initiators review for FLEX/portable equipment) in the next model update. FLEX has been shown to have a limited impact on PRA results and on system and component categorization results (See response to RAI APLA 4a).

This FLEX 10 CFR 50.69 sensitivity is documented in a PRA report. This report and the results will be shared with the 10 CFR 50.69 Integrated Decision-making Panel (IDP) to ensure they are aware of the impact FLEX modeling has on PRA results and importance measures. The potential inclusion of pre-initiator events in FLEX treatment is bounded by sensitivity case (baseline model with FLEX and no pre-initiators compared to zero credit for FLEX equipment/actions).

b) ii b) The documented sensitivity case and resulting importance measure comparison provides the range of results that bound inclusion of pre-initiating events. The baseline case is likely slightly less conservative than a model with pre-initiators and the comparison case has no credit at all for FLEX. Given that FLEX is considered a key source of uncertainty for this application, this sensitivity study will be shared with the Integrated Decision-making Panel (IDP) for categorization of the specified systems to ensure they are aware of the impact FLEX modeling has on PRA results and importance measures.

b) iii a) The procedural entry into the implementation of FLEX equipment or mitigating strategies at Waterford is clear and explicit. The Station Blackout (SBO) emergency operating procedure OP-902-005 directs operators to enter the FLEX strategy for any event involving total loss of AC power in which power is not expected to be restored within the License Basis SBO timeframe of 4 hours. This decision must be made within an hour of the onset of total loss of AC power. Additionally, for situations where SBO is not the initiating event, but loss of additional equipment induces ELAP conditions with the same timing considerations, then operators are routed to the FLEX strategy guidelines via the SBO procedure. The initial cues for operators are obvious and not ambiguous. This has been confirmed through discussions with operations personnel.

In preliminary FLEX HEP development, ELAP declaration was considered an initial cue for the Deep Load Shed event and was not considered a separate/unique human action. Upon review, it appears the ELAP declaration should be included in the model as a sperate HEP event. A Waterford PRA MCR has been issued to track this technical issue and ensure it is addressed in the next PRA model update.

b) iii b) The FLEX sensitivity case shows that there will be no variation in the population of candidate safety significant components based on FLEX modeling treatments when RAW and FV importance measures are compared. The only exception to this would be if EFW and/or EDG systems (and systems that directly support them) are included in categorization efforts. Based on the critical safety function and high-risk importance of EFW and EDG systems, categorization of the impacted components is unlikely.

The FLEX 10 CFR 50.69 sensitivity case (see response to APLA 04a) shows the limited impact FLEX treatment has on potential 10 CFR 50.69 categorization. Given that this item is a key source of uncertainty for this application, this report and the results will be shared with the 10 CFR 50.69 Integrated Decision-making Panel (IDP) for categorization of the specified systems to ensure they are aware of the impact FLEX modeling has on PRA results and importance measures. The technical issues associated with ELAP declaration and modeled FLEX HEP events is bounded by the sensitivity results. The IDP will have the list of impacted components and systems to allow for informed decision-making for all 10 CFR 50.69 categorization efforts.

c) i When FLEX was added to the model originally, the HRA events were included in the dependency analysis. The next model update (the current model) removed them and provided a justification for excluding them in the HRA documentation. It was later identified that the basis for the dependency exclusion was insufficient. A PRA MCR was created to ensure the FLEX HEP events are included in the dependency evaluation in the next PRA model update. An immediate fix or revision was judged to be unnecessary due to the limited impact of FLEX and even smaller impact the dependency change would have on results.

While the current model does not represent the ideal dependency treatment of these events, this was not corrected immediately and was judged to not significantly impact results. The following points supported the independent modeling treatment.

- These events are relatively independent.
- Significantly different timing with FLEX actions having time delays of 60-120 minutes and most other SBO (FLEX is only credited in SBO sequences) related actions needing to be completed in under an hour.

The FLEX 10 CFR 50.69 sensitivity case (see response to APLA 04a) show the limited impact FLEX treatment has on potential 10 CFR 50.69 categorization. This report and the results will be shared with the 10 CFR 50.69 Integrated Decision-making Panel (IDP) to ensure they are aware of the impact FLEX modeling has on PRA results and importance measures. The HEP dependency treatment for FLEX events has a very small impact on PRA results and likely no impact on 10 CFR 50.69 categorization results.

c) ii. The HRA Calculator Dependency Decision Tree tool was used in HRA development for the Revision 6 Waterford 3 PRA model. This is documented in the HRA documentation for the PRA model (PSA-WF3- 01-HR). However, the FLEX events were excluded from this during this modeling development.

c) iii. The FLEX 10 CFR 50.69 sensitivity case (see response to APLA 04a) show the limited impact FLEX treatment has on potential 10 CFR 50.69 categorization. Given that credit for FLEX is a key source of uncertainty relative to the EDG, EFW, and FLEX systems, the results of this sensitivity study will be shared with the Integrated Decision-making Panel (IDP) for categorization of these systems to ensure they are aware of the impact FLEX modeling has on PRA results and importance measures. The HEP dependency treatment for FLEX events has a very small impact on PRA results and likely no impact on 10 CFR 50.69 categorization results.

Enclosure 2

W3F1-2021-0050

Responses to APLC RAIs

Waterford 3 50.69 LAR

Responses to APLC RAIs

By letter dated December 18, 2020 (Agencywide Documents Access and Management System (ADAMS) Accession No. ML20353A433), Entergy Operations, Inc (Entergy or the licensee) submitted a license amendment request (LAR or the application) for the use of a risk-informed process for the categorization and treatment of structures, systems, and components at Waterford Steam Electric Station, Unit 3 (Waterford). The proposed license amendment would modify the Waterford licensing basis, by the addition of a License Condition, to allow for the implementation of the provisions of Title 10 of the Code of Federal Regulations (CFR), Part 50.69, "Risk-Informed Categorization and Treatment of Structures, Systems and Components for Nuclear Power Reactors." The U.S. Nuclear Regulatory Commission (NRC) staff from Probabilistic Risk Assessment Licensing Branch C (APLC) has reviewed the LAR and requests additional information (RAI) in order to complete the review.

APLC Question 01 – Alternative Seismic Approach

Section 50.69(b)(2)(ii) of 10 CFR requires that the quality and level of detail of the systematic processes that evaluate the plant for external events during operation are adequate for the categorization of systems, structures and components (SSCs).

In the Waterford 3 license amendment request (LAR), the licensee proposes to address seismic hazard risk using the alternative seismic approach for seismic Tier 1 plants described in EPRI Report 3002017583 ("Alternative Approaches for Addressing Seismic Risk in 10 CFR 50.69 Risk-Informed Categorization," dated February 11, 2020) and other qualitative considerations. The NRC staff understands that EPRI Report 3002017583 is an updated version of EPRI Report 3002012988 ("Alternative Approaches for Addressing Seismic Risk in 10 CFR 50.69 Risk-Informed Categorization," dated July 2018), which was reviewed in conjunction with the NRC staff's review of the Calvert Cliffs Nuclear Power Plant (Calvert Cliffs) Units 1 and 2 LAR for adoption of 10 CFR 50.69, dated November 28, 2018 (ADAMS Accession No. ML18333A022). Calvert Cliffs was the pilot plant for using the alternative seismic Tier 1 approach described in EPRI Report 3002012988. The NRC staff has not endorsed EPRI Report 3002012988 as a topical report for generic use. As such, each licensee is required to perform a plant-specific evaluation of applicability of the EPRI alternative seismic approach to their plant.

The NRC staff reviewed and approved the Calvert Cliffs alternative seismic Tier 1 approach based on the information on Tier 1 plants included in EPRI Report 3002012988 and the information provided in the supplements to the Calvert Cliffs LAR. Information in the supplements to the Calvert Cliffs LAR (ADAMS Accession Nos. ML19130A180, ML19183A012, ML19200A216, and ML19217A143) that was used to support the NRC staff's review and approval of the Calvert Cliffs alternative seismic Tier 1 approach is included in the NRC staff's safety evaluation for the Calvert Cliffs LAR (ADAMS Accession No. ML19330D909). The NRC staff notes that the licensee's proposed alternative seismic Tier 1 approach is similar to that reviewed and approved in the NRC staff's Calvert Cliffs safety evaluation. However, the licensee's approach for Waterford 3 is based on EPRI Report 3002017583 instead of EPRI Report 3002012988. Please address the following:

a. Since EPRI Report 3002017583 is cited in the LAR; the report should be submitted on the docket for NRC staff review.

b. Explain whether the information in EPRI Report 3002012988 and in the supplements to the Calvert Cliffs LAR used to support the NRC staff's review and approval of the Calvert Cliffs alternative seismic Tier 1 approach are fully represented in EPRI Report 3002017583 and the LAR for Waterford 3. If there are any gaps between the two sets of information, any missing information should be identified and incorporated into the Waterford 3 LAR, as applicable.

c. Identify and justify differences, if any, between the Waterford 3 proposed alternative seismic Tier 1 approach and that reviewed and approved in the NRC staff's Calvert Cliffs safety evaluation, including any Waterford 3-specific considerations.

Waterford 3 Response

a. The requested EPRI document is provided via a secured portal to the NRC staff for review.

b. The information in EPRI Report 3002012988 and in the supplements to the Calvert Cliffs LAR (ML19183A012, ML19200A216, and ML19217A143) used to support the NRC staff's review and approval of the Calvert Cliffs alternative seismic Tier 1 approach are fully represented in EPRI Report 3002017583. This report is provided in Part a. of this response.

The technical criteria in EPRI Report 3002017583 are unchanged from EPRI Report 3002012988. The Product Description at the beginning of EPRI Report 3002017583 states the following:

> "This Technical Update incorporates updates submitted to the NRC in an RAI submittal for the Calvert Cliffs 50.69 LAR into the previous version of this report, EPRI 3002012988. Aside from those updates, the technical criteria in this report remains unchanged."

Calvert Cliffs' July 19, 2019 RAI response (ML19200A216 – cited in this APLC Question 01) provided the seismic alternative markups to Report 3002012988.

In addition, EPRI Report 300217583 incorporated a few minor editorial changes including the following:

1. Figure 1-2 was edited to include EPRI 3002017583 in the list of 10 CFR 50.69 supplemental guidance documents
2. Figure 2-2, Low Seismic Hazard Site: Typical SSE to GMRS comparison replaced graph with correct graph.

EPRI 3002017583 has incorporated all the information and follow up actions from the CCNPP LAR supplements that was agreed upon by the NRC staff's review of the alternative seismic approach for Tier -1 plants. Therefore, Attachment 2 of ML19200A216 is applicable to Waterford 3 since it is using the updated EPRI Document 3002017583, Alternative Approaches for Addressing Seismic Risk in 10 CFR 50.69 Risk-Informed Categorization.

c. In review of the CCNPP SE, one difference was identified from the proposed alternative seismic approach documented in the Waterford 3 LAR. As discussed below, this will be incorporated into the program and categorization process.

In the section "Monitoring of Inputs to and Outcome of Proposed Alternative Seismic Approach" of the CCNPP SE, the configuration control program for CCNPP had been updated to include a checklist of configuration activities to recognize those systems that have been categorized in accordance with 10 CFR 50.69, to ensure that any physical change to the plant or change to plant documents is evaluated prior to implementing those changes. This checklist is the same as what is included in Section 3.5 of the Waterford 3 LAR except for "Review of impact to seismic loading and SSE seismic requirements, as well as the method of combining seismic components." This checklist item will also be included in the Entergy configuration control program.

APLC Question 02 – External Hazards Screening

NEI 00-04 (ADAMS Accession No. ML052910035), Revision 0, Section 5.4, provides guidance on assessment of other external hazards (excluding fire and seismic) in 10 CFR 50.69 categorization of SSCs. Specifically, Figure 5-6, "Other External Hazards," in NEI 00-04 illustrates a process that begins with an SSC selected for categorization and proceeds through a flow chart for each external hazard. Figure 5-6 of NEI 00-04 shows that if a component participates in a screened scenario then, in order for that component to be considered as a low safety significant (LSS) item, it has to be further shown that if the component was removed the screened scenario would not become unscreened. NEI 00-04 explicitly states, in part, that "[i]f it can be shown that the component either did not participate in any screened scenarios or, even if credit for the component was removed, the screened scenario would not become unscreened, then it is considered a candidate for the low safety-significant category."

Section 3.2.4, "Other External Hazards," of the Waterford 3 LAR Enclosure states, in part, "[a]ll external hazards, except for seismic, were screened from applicability to Waterford 3 per a plant-specific evaluation in accordance with Generic Letter (GL) 88-20 ('Individual Plant Examination of External Events (IPEEE) for Severe Accident Vulnerabilities – 10 CFR 50.54(f),' Supplement 4, dated June 28, 1991 (ADAMS Accession No. ML031150485)) and updated to use the criteria in the ASME/ANS PRA Standard RA-Sa-2009." Attachments 4 and 5 of the LAR Enclosure address the results of other external hazards screening and the progressive screening approach, respectively. However, the licensee does not address any considerations with respect to the application of Figure 5-6 of NEI 00-04 to the screening of other external hazards at Waterford 3. Please address the following:

a. Clarify whether SSCs credited for screening of external hazards will be evaluated using the guidance illustrated in Figure 5-6 of NEI 00-04 during the implementation of the 10 CFR 50.69 categorization process at Waterford 3.

b. Identify the external hazards addressed in Attachment 4, "External Hazards Screening," of the LAR Enclosure that will be evaluated according to the flowchart in Figure 5-6 of NEI 00-04.

c. If the approach illustrated in Figure 5-6 of NEI 00-04 will not be used, describe the Waterford 3 proposed approach and provide its justification.

d. Attachment 4 to the LAR Enclosure indicates that the tornado missile hazard is screened based on a recent tornado hazard analysis. It is unclear to the NRC staff if the analysis included the assessment of NRC Regulatory Issue Summary (RIS) 2015-06, "Tornado Missile Protection," dated June 10, 2015 (ADAMS Accession No. ML15020A419).

    i. Clarify whether the recent analysis included the RIS 2015-06 assessment.

    ii. Provide justification, as applicable, that any non-conformances identified in the assessment do not impact the screening of tornado missile hazard.

    iii. Alternatively to Part ii, provide an updated screening analysis for the extreme wind and tornado hazard.

<u>Waterford 3 Response</u>

a. SSCs credited for screening of external hazards will be evaluated according to the flow chart in NEI 00-04, Figure 5-6.

b. All External Hazards listed in Attachment 4 "External Hazards Screening" will be evaluated according to the flowchart in Figure 5-6 of NEI 00-04. This figure provides the NRC approved process to be used to determine SSC safety significance for other external hazards (excluding internal fires and seismic hazards). Waterford 3 is following NEI 00-04 Section 5.4 for assessment of other external hazards. As part of the categorization assessment of "other external hazard" risk, an evaluation is performed to determine if there are components being categorized that participate in screened scenarios and whose failure would result in an unscreened scenario. Such components would be categorized as high safety significant "HSS".

c. SSCs credited for screening of external hazards will be evaluated according to the flow chart in NEI 00-04, Figure 5-6.

d.i. Upon receipt of RIS 2015-06, Waterford completed an evaluation to determine if Waterford 3 was susceptible to the issues identified in RIS 2015-06 (CR-WF3-2015-05956). This assessment reviewed the following site documents:

- TORMIS Analysis: Tornado Generated Missile Strike at WF3 (ECC-99-008)

- WF3 Tornado Basis Design Criteria (W3-CS-98-001-00)

- Safe Shutdown Analysis

The conclusion of the assessment was that Waterford has no known degraded or non-conforming barriers. Though no degraded or non-conforming barrier were identified, the Tornado Design Basis Report (W3-CS-98-001-00) was revised to include required safe shutdown equipment to ensure the analysis was more robust for the RIS 2015-06 issues.

The recent analysis referenced in Attachment 4 of the LAR did not explicitly consider RIS-2015-06. The LAR referenced High Wind assessment was a review of the current Waterford 3 licensing basis tornado and extreme winds against updated extreme wind and tornado data from ASCE 7-10, RG 1.76 Rev 3, and NUREG/CR-4461 Rev. 2. With no known degraded or non-conforming conditions/barriers, assessment of RIS 2015-06 was not necessary as it had been dispositioned previously.

d.ii. The site-specific review documented that no non-conformances were identified or present and therefore have no impact on the screening of this hazard.

d.iii. No additional analysis is required for screening extreme wind and tornado hazards.

APLC Question 03 – Seismic Risk Contribution

Section 50.69(b)(2)(ii) of 10 CFR requires that a LAR include a "description of the measures taken to assure that the quality and level of detail of the systematic processes that evaluate the plant for internal and external events during normal operation, low power, and shutdown … are adequate for the categorization of SSCs." Section 50.69(b)(2)(iv) of 10 CFR requires that a LAR include a "description of, and basis for acceptability of, the evaluations to be conducted to satisfy 10 CFR 50.69(c)(1)(iv)." The Statement of Consideration (SOC) on 10 CFR 50.69(b)(2)(iv) of the Final rule published in the Federal Register on November 22, 2004 (69 FR 68008)) states that the licensee, "is required to include information about the evaluations they intend to conduct to provide reasonable confidence that the potential increase in risk would be small." The SOC further clarifies that a "licensee must provide sufficient information to the NRC, describing the risk sensitivity study and other evaluations and the basis for their acceptability as appropriately representing the potential increase in risk from implementation of the requirements in the rule."

In Section 3.2.3, "Seismic Hazards," of the LAR Enclosure, the licensee states, in part, that "low seismic CDF and LERF estimates lead to reasonable confidence that seismic risk contributions would allow reducing an HSS to LSS due to the 10 CFR 50.69 Integral Assessment if the equipment is HSS only due to seismic considerations." Section 2.2.2 of EPRI Report 3002017583 identifies the contribution of seismic to total plant risk as a basis for the use of the proposed alternative seismic approach for Tier 1 sites. However, the NRC staff notes that the LAR does not provide information to show that the plant-specific seismic risk constitutes a small fraction of the total plant risk and thus that the proposed alternative seismic approach is applicable to Waterford 3.

In Section 3.2.3 of the LAR Enclosure, the licensee further states that "Waterford 3 completed a bounding seismic risk evaluation to support development of a Risk-Informed Completion Time (TSTF-505) license amendment request and program." Based on the Technical Specifications Task Force (TSTF) Traveler TSTF-505 LAR (ADAMS Accession No. ML21039A648) for Waterford 3, it appears that the seismic penalty was based on a plant's high-confidence of low-probability of failure (HCLPF) capacity of 0.25g, as opposed to 0.1g in the Generic Issue 199 report (Safety/Risk Assessment Results for Generic Issue 199, "Implications of Updated Probabilistic Seismic Hazard Estimates in Central and Eastern United States on Existing Plants," dated September 2, 2010 (ADAMS Accession No. ML100270582)).

a. Provide justification for a plant HCLPF capacity of 0.25g used for Waterford 3.

b. Justify that the plant-specific seismic risk is low relative to the overall plant risk such that the categorization results will not be significantly impacted to support the applicability of the proposed alternative seismic approach to Waterford 3.

<u>Waterford 3 Response</u>

a. The Waterford Unit 3 (WF3) TSTF-505 License Amendment Request (LAR), referenced in the WF3 50.69 LAR, inappropriately cited 0.25g Peak Ground Acceleration (PGA) as the WF3 plant level fragility high-confidence of low-probability of failure (HCLPF) value cited by the NRC in the Generic Issue 199 risk assessment report. WF3 PRA report PSA-WF3-04-01 was revised to address this point and to provide justification of the reasonableness of 0.25g WF3 plant level fragility HCLPF in the following manner:

Justification of reasonableness of 0.25g WF3 plant level fragility HCLPF

- General aspects of seismic margin assessment (SMA)-based plant level seismic fragilities were assessed to demonstrate that the 0.1g WF3 plant level fragility HCLPF cited by the NRC in the Generic Issue 199 risk assessment report was an evaluation constraint of the IPEEE program and the result of the WF3 IPEEE SMA should not be viewed as a reasonable estimate of the WF3 plant level fragility HCLPF given the very low seismic margins earthquake (i.e., the SSE) used.
- The range of plant level fragilities from Near Term Task Force (NTTF) 2.1 Seismic PRAs was reviewed to establish that plant level seismic HCLPFs are typically in the range of 0.2g to 0.3g PGA (or higher) and that a 0.25g PGA HCLPF value is reasonable for WF3.
- A review of the margin in the WF3 plant beyond the design basis Safe Shutdown Earthquake (SSE) was performed considering the WF3 seismic design basis along with additional seismic evaluation work that has been performed at WF3 to identify likely sources of seismic margin. The review considers seismic walkdown efforts, WF3 seismic demand, and seismic capacity considerations including the use of SMA experience data, as well as seismic interactions, relay chatter, and equipment anchorage.

The review detailed in PSA-WF3-04-01 was performed to support the assertion that 0.25g is reasonable for use as a plant level seismic HCLPF for WF3. Seismic walkdowns confirm that WF3 includes good seismic design and does not have issues with significant seismic interactions. A variety of conservatisms, in terms of both seismic demand and seismic capacity, were identified and catalogued and indicate seismic margin beyond design-basis. While it is not possible to quantify a specific plant-level HCLPF since Waterford 3 does not have a seismic PRA, sufficient seismic margin has been identified to judge that a 0.25g plant-level HCLPF is reasonable. The revised report is included in Attachment 1 of this enclosure.

b. The following information is provided to support that seismic risk will not solely result in a high safety significance (HSS) determination based on integrated importance measures and therefore, will not challenge the use of the qualitative consideration of seismic risk in the proposed approach.

Seismic Penalty Values Not Bounding

The WF3 50.69 LAR and WF3 TSTF-505 LAR inappropriately characterized the RICT seismic penalty estimates themselves as "bounding". These values are not bounding, as described below. Similar clarifications that seismic penalty SCDF and SLERF values are not "bounding" have been made in other docketed responses (refer to NRC ADAMS Accession Nos. ML20329A433 and ML20337A301).

Throughout the risk managed technical specifications guideline NEI 06-09 Rev 0-A and the NRC SE for that document, reference is made to either a "bounding" or "conservative" analysis, or sometimes to a "reasonable bounding analysis", as being acceptable to account for risk for external hazards when a PRA model is not available. The references to estimation of a seismic CDF or a seismic LERF contribution for the RICT program as "bounding" is typically inappropriate, and in the case of the WF3 TSTF-505 LAR it is inappropriate. One approach to a "bounding" estimate for SCDF would be the annual exceedance frequency of the SSE (i.e., assume a conditional core damage probability of 1.0 for SSE loads) and a "bounding" approach for estimation of SLERF would be to assume a seismic conditional LERF probability (SCLERP) of 1.0 (i.e., SCDF and SLERF are equal to the exceedance frequency of the SSS). Neither of these "bounding" approaches are reasonable or useful for risk managed technical specifications.

The estimation of seismic risk results for the WF3 RICT program are more appropriately characterized as a "conservative" analysis that uses the following:

- Estimated SCDF
- Estimated average SCLERP to determine an estimated SLERF, and
- Conservative implementation of the SCDF and SLERF used in RICT assessments.

Estimated SCDF

The WF3 seismic penalty evaluation uses an estimate of SCDF and this estimate is not bounding but is a nominal estimate with some conservative aspects in its calculation. As discussed in PSA-WF3-04-01, the SCDF value is determined from a mathematical convolution of the WF3 plant seismic hazard curve and an estimate of the WF3 plant level seismic fragility curve.

Estimated SCLERP

The WF3 SLERF seismic penalty is calculated by multiplying the SCDF convolution estimate by an average seismic conditional LERF probability (SCLERP) of 0.15. This is an updated value from the 0.10 SCLERP in the previous analysis. The reasonableness of the 0.15 SCLERP assumed in the WF3 seismic penalty calculation is discussed in PSA-WF3-04-01 considering the following points: Breakdown of SCDF by accident sequence type, and Containment Isolation.

Breakdown of SCDF by Accident Sequence Type

A given accident sequence type may not result in a core damage event until well after the PRA "Early" release time frame. Conversely, some accident sequence types would, by PRA convention, be modeled directly as a LERF release, such as seismic-induced containment failure or bypass. The contribution of various accident sequence types (or accident classes) to core damage frequency at a given plant is not necessarily the same between FPIE PRA and other hazard (e.g., seismic) PRAs.

Further evidence regarding the reasonableness of the SCLERF estimate is presented in PSA-WF3-04-01.

1. <u>WF3 Hazard Meets EPRI 3002017583 Tier 1 Criteria:</u> As discussed in Section 3.2.3 of the WF3 50.69 LAR, the WF3 2014 seismic hazard (NRC ADAMS Accession No. ML14086A427) meets the low hazard (Tier 1) criteria specified in EPRI 3002017583. As stated in the LAR, *"at these sites, the GMRS is either very low or within the range of the SSE such that unique seismic categorization insights are not expected."*
2. <u>Limited Unique Seismic Insights:</u> The NRC is correct that Section 3.2.3 of the WF3 50.69 LAR includes statements that imply that low estimated seismic risk is key to use of the EPRI 3002017583 Tier 1 seismic alternative process. However, the EPRI seismic alternative report 3002017583 (Section 2.2.2.1) does not explicitly state that the relative contribution of seismic risk is (or needs to be) low compared with the overall plant risk to justify application of the Tier 1 seismic alternative process. In fact, based on the trial studies, EPRI 3002017583 concludes that for both low hazard sites and higher hazard sites that the potential of an SSC being identified as HSS uniquely due to seismic risk calculations is low likelihood:

   "The test cases described in Section 3 showed that even for plants with high seismic ground motions compared to their design basis, there would be very few if any SSCs designated HSS for seismic unique reasons. At the low seismic hazard sites in Tier 1, the likelihood of identifying a unique seismic condition that would cause an SSC to be designated HSS is very low." (Section 2.2.2 [5])

3. <u>Conservatisms in SCDF and SLERF Penalty Estimates:</u> As discussed previously, the SCDF and SLERF penalty estimates are approximations containing aspects of conservatism. In addition, the SCDF and SLERF seismic penalty approximations do not explicitly account for the following diverse accident mitigation features in place at the WF3 plant:

   - TEDG: The temporary emergency diesel generator (TEDG) provides sufficient power capacity for safe plant shutdown as each of the safety related emergency diesel generators, but the TEDG is diverse in design and location. This diversity in design and location means that it would not be postulated to experience a seismic-induced correlated failure (i.e., increased likelihood of all EDGs failing due to similar seismic capacity and seismic response) with the other original design diesel generators. The TEDG is a skid-mounted unit that has been used during refueling outages but it has been made a permanent modification (now credited in the WF3 FPIE and FPRA PRA models). TEDG is permanently located onsite and can be aligned to either safety division.
   - FLEX: FLEX mitigating strategies were developed in response to the Fukushima accident. The FLEX strategies are designed to reduce the risk contribution for beyond design basis scenarios. FLEX "N" equipment at WF3 is located entirely with the Reactor Aux Building (RAB), which is a seismically qualified structure. The FLEX Diesel (and associated fuel tank) is designed to meet the criteria of seismically "robust" per NEI 12-06. The FLEX Core Cooling pump and required hoses and cables are mounted according to seismic II/I criteria. These components represent the primary FLEX

components credited in the WF3 PRA. The FLEX strategy relies primarily on permanently installed equipment with portable fluid and power connections being implemented in response to the accident. These connections are not rigid and consist of portable cable reels and flexible hoses that are stored in close proximity to the component they support.

4. Significant Fraction of SCDF and SLERF Estimate Not Directly Applicable: Although not explicitly addressed and accounted for in the NEI 00-04 construct for SSC categorization, a significant fraction of calculated SCDF and SLERF would not be directly applicable or useful for SSC categorization purposes. This fraction is comprised of seismic induced severe damage states (e.g., seismic-induced building failures, seismic-induced RPV support failure, seismic-induced containment failure, seismic-induced containment bypass) that are modeled in seismic risk assessments as leading directly to SCDF (and also directly to SLERF for some portion of those severe damage states). Non-seismic failure modes (e.g., pump fails to run or start, pump in test or maintenance, valve fails to change position) are as a general rule non-significant contributors to SCDF or SLERF.

APLC Question 04 – <u>Change of Seismic Hazards</u>

Regulatory Position C.9, "NRC Endorsement of Revision 0 of NEI 00-04; Specific Clarifications," of RG 1.201, Revision 1, "Guidelines for Categorizing Structures, Systems, and Components in Nuclear Power Plants According to their Safety Significance," dated May 2006 (ADAMS Accession No. ML061090627), states, in part:

> "As part of the NRC's review and approval of a licensee's or applicant's application requesting to implement §50.69, the NRC staff intends to impose a license condition that will explicitly address the scope of the PRA and non-PRA methods used in the licensee's categorization approach. If a licensee or applicant wishes to change its categorization approach and the change is outside the bounds of the NRC's license condition (e.g., switch from a seismic margins analysis to a seismic PRA), the licensee or applicant will need to seek NRC approval, via a license amendment, of the implementation of the new approach in their categorization process."

In Section 3.2.3 of the LAR Enclosure, the licensee states:

> "In the unlikely event that the Waterford 3 seismic hazard changes to medium risk (i.e., Tier 2) at some future time, Waterford 3 will follow its categorization review and adjustment process procedures to review the changes to the plant and update, as appropriate, the SSC categorization in accordance with 10 CFR 50.69(e)."

It appears this statement indicates that the licensee will switch to the Tier 2 approach, which is outside of this proposed alternative seismic Tier 1 approach, without prior review and approval by the NRC staff.

Confirm that the licensee will seek prior NRC approval if the licensee's feedback process determines that a process different from the proposed alternative seismic Tier 1 approach is warranted for seismic risk consideration in categorization under 10 CFR 50.69

Waterford 3 Response

Entergy will seek prior NRC approval if the site determines that a process different from the proposed alternative seismic Tier 1 approach is warranted for seismic risk consideration in categorization under 10 CFR 50.69.

Enclosure 2, Attachment 1

W3F1-2021-0050

PSA-WF3-04-01

(59 pages follow)

Entergy

ENTERGY NUCLEAR
Engineering Report Cover Sheet

Engineering Report Title:
Waterford 3
Seismic Risk Evaluation to Support the TSTF-505 LAR

Engineering Report Type: (3)

New ☐ Revision ☒ Cancelled ☐ Superseded ☐
Superseded by: ____________

Applicable Site(s) (4)

IP1	☐	IP2	☐	IP3	☐	JAF	☐	PNPS	☐	VY	☐	WPO	☐
ANO1	☐	ANO2	☐	ECH	☐	GGNS	☐	RBS	☐	WF3	☒	PLP	☐

EC No. 90883

Report Origin: ☒ Entergy ☐ Vendor
Vendor Document No.: ____________

Quality-Related: ☐ Yes ☒ No

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Waterford 3

Seismic Risk Evaluation to Support the TSTF-505 LAR

PSA-WF3-04-01, Revision 2

Non-Quality Related

Table of Contents

List of Tables

1. Introduction

NEI-06-09 guidelines (Ref. 6) require TSTF-505 LAR submissions to justify excluding any risk sources determined to be insignificant to the calculation of configuration-specific risk, and also to provide a discussion of any conservative or bounding analyses to be applied to the calculation of risk-informed completion times (RICTs) for sources of risk not addressed by the PRA models.

Waterford 3 seismic risk was assessed using a Seismic Margins Analysis. As a result, no seismic PRA is available to determine seismic contribution to Risk Informed Completion Times.

To support the Waterford 3 (W3) TSTF-505 LAR submission, this document examines the potential to exclude seismic hazard from the calculation of configuration-specific risk and provides a conservative analysis to be applied to the calculation of risk-informed completion times (RICTs).

2. Review of Seismic Risk Estimates

W3 seismic risk is not addressed by a PRA model. For the individual plant examination of external events (IPEEE), a screening-level[1] seismic margins assessment was performed. The W3 TSTF-505 LAR submission will therefore need to justify excluding seismic risk as insignificant to the calculation of configuration-specific risk or apply a reasonable analysis to the calculation of RICTs. This section performs a review of seismic risk estimates to determine whether seismic risk can be screened from the W3 TSTF-505 LAR submission.

2.1 Published Seismic Risk Estimates

The USNRC recently published information on the estimates of the seismic risk levels for all plants in the Central and Eastern United States (CEUS) as part of Generic Issue 199 (Ref. 1). In order to address the changing state of knowledge on seismic hazards, the NRC Staff developed a technical analysis that computed conservative estimates of seismic risk for all plants in the CEUS (Ref. 2). The technical analysis uses a variety of calculation approaches to compute a conservative estimate of the seismic core damage frequency (SCDF) using three different seismic hazard sources. The results for W3 are provided in Table 2-1.

[1] Three W3 seismic screening and verification walkdowns were performed. The purpose of the walkdowns was to assess the relative seismic capacity of plant structures systems and components. Two plant conditions were identified that warranted modification packages, and these were implemented. There were no other issues identified that necessitated additional changes to plant operations and procedures to ensure seismic adequacy of W3.

Table 2-1 Published Estimates of Waterford 3 Seismic Core Damage Frequency

	SCDF Calculation Approach				
Hazard Source	**Maximum Spectral Result (/yr)**	**Simple Average (/yr)**	**IPEEE Weighted Average (/yr)**	**Weakest Link Model (/yr)**	**Highest Estimate (/yr)**
1989 EPRI (Ref. 3)	1.2E-06	7.3E-07	6.7E-07	1.2E-06	1.2E-06
1994 LLNL (Ref. 4)	2.9E-05	1.9E-05	1.8E-05	2.9E-05	2.9E-05
2008 USGS (Ref. 5)	1.8E-05	1.3E-05	1.2E-05	2.0E-05	2.0E-05

The maximum W3 SCDF value generated is 2.9E-05/yr, based on the 1994 LLNL hazard. All SCDF estimates are greater than the 1E-06/yr bounding core damage frequency for beyond design basis hazard conditions (Ref. 8).

2.2 Seismic Risk Estimate Based on the 2014 Updated Seismic Hazard

A W3 seismic hazard and screening report was submitted in 2014 in response to Fukushima Near-Term Task Force (NTTF) Recommendation 2.1 (Ref. 7). The results of the screening report determined that an additional seismic risk evaluation was not required. For the 1 to 10 Hz part of the response spectrum, the screening evaluation indicates that the safe shutdown earthquake (SSE) envelopes the ground motion response spectrum (GMRS). For the high frequency portion of the response spectrum, the GMRS accelerations in the high frequency range and frequency content above 10 Hz are filtered out by the soil-structure system and no further considerations are required.

To examine whether seismic events can be excluded from the calculation of configuration-specific risk, a seismic CDF was estimated based on the updated W3 seismic hazard using the methodology discussed in Reference 12, Appendix 10-B. First, a plant high-confidence low-probability of failure (HCLPF) was obtained for seismic capacity. The plant level HCLPF is assumed to be 0.25g based on the evaluation in Appendix A. Assuming commonly used fragility uncertainty parameter value of $\beta_c = 0.4$, the median capacity of the plant is A_m = 0.63g.

$$A_m = \text{HCLPF} / e^{-2.33(\text{Bc})} = 0.25g / e^{-2.33(0.4)} = 0.63g$$

Using the FRANX software code (Ref. 10), a convolution was performed over the full range of the PGA hazard curve for A_m = 0.63g, Bc = 0.4. The hazard frequencies and ground acceleration values were obtained from Table A-1d of Reference 7. To adjust the PGA-based A_m and HCLPF for the 1, 5 and 10 Hz Spectral Accelerations per Generic Issue 199, Appendix C (Ref. 1); the ratio of the acceleration at the 100 Hz value and the frequency of interest is applied to the PGA median seismic capacity (A_m) value.

The FRANX scenarios were set up with 8 bins to cover the range of seismic accelerations. The lowest range starts at the Operating Basis Earthquake (OBE) which, for Waterford, is the same as the SSE (0.1g). The bins were divided evenly into 0.2g increments up to 1.5g with the 8th bin being all accelerations >1.5g. Since the convolution is being performed using a plant level fragility, the number of bins is not important. Eight bins provides a reasonable distribution for this analysis.

The hazard data was copied from ML14086A427 (Ref. 7). The applicable columns of data for each hazard curve (AMPS, 0.16, Median, Mean and 0.84) were copied into the correct format and then pasted into the FRANX Hazard Editor. One FRANX file was created for each Hazard Curve (1 Hz, 5 Hz, 10 Hz and PGA).

This convolution produces a highest SCDF of 1.26E-06/yr, as summarized in Table 2-2 through 2-5. Three of the four estimates are greater than the 1E-06/yr bounding core damage frequency for beyond design basis hazard conditions (Ref. 8).

Table 2-2 Convolution of the 0.25g W3 Plant HCLPF over the Range of the PGA Hazard Curve

Scenario	Description	Earthquake Frequency (/yr)	CCDP	SCDF (/yr)
%G01	%G01 - Hazard Curve: W3 PGA Hazard - PGA Range: 0.1g to 0.3g	5.42E-05	6.24E-04	3.38E-08
%G02	%G02 - Hazard Curve: W3 PGA Hazard - PGA Range: 0.3g to 0.5g	3.92E-06	1.12E-01	4.39E-07
%G03	%G03 - Hazard Curve: W3 PGA Hazard - PGA Range: 0.5g to 0.7g	8.48E-07	4.38E-01	3.71E-07
%G04	%G04 - Hazard Curve: W3 PGA Hazard - PGA Range: 0.7g to 0.9g	2.76E-07	7.18E-01	1.98E-07
%G05	%G05 - Hazard Curve: W3 PGA Hazard - PGA Range: 0.9g to 1.1g	1.10E-07	8.73E-01	9.58E-08
%G06	%G06 - Hazard Curve: W3 PGA Hazard - PGA Range: 1.1g to 1.3g	5.33E-08	9.45E-01	5.04E-08
%G07	%G07 - Hazard Curve: W3 PGA Hazard - PGA Range: 1.3g to 1.5g	2.75E-08	9.77E-01	2.69E-08
%G08	%G08 - Hazard Curve: W3 PGA Hazard - PGA Range: > 1.5g	4.59E-08	9.92E-01	4.55E-08
			Total	**1.26E-06**

To adjust the 5 Hz:
ratio = 0.207/0.11
A_{m5} = 0.635*0.207/0.11 = 1.195
$HCLPF_5$ = 0.470

Table 2-3 Convolution of the WF3 Plant HCLPF over the Range of the 5 Hz Hazard Curve

Scenario	Description	Earthquake Frequency (/yr)	CCDP	SCDF (/yr)
%G01	%G01 - Hazard Curve: W3 5 Hz Hazard - Range: 0.1g to 0.3g	2.16E-04	6.90E-07	1.49E-10
%G02	%G02 - Hazard Curve: W3 5 Hz Hazard - Range: 0.3g to 0.5g	1.63E-05	2.43E-03	3.97E-08
%G03	%G03 - Hazard Curve: W3 5 Hz Hazard - Range: 0.5g to 0.7g	4.00E-06	3.94E-02	1.58E-07
%G04	%G04 - Hazard Curve: W3 5 Hz Hazard - Range: 0.7g to 0.9g	1.43E-06	1.53E-01	2.19E-07
%G05	%G05 - Hazard Curve: W3 5 Hz Hazard - Range: 0.9g to 1.1g	5.93E-07	3.24E-01	1.92E-07
%G06	%G06 - Hazard Curve: W3 5 Hz Hazard - Range: 1.1g to 1.3g	2.87E-07	5.01E-01	1.44E-07
%G07	%G07 - Hazard Curve: W3 5 Hz Hazard - Range: 1.3g to 1.5g	1.43E-07	6.52E-01	9.30E-08
%G08	%G08 - Hazard Curve: W3 5 Hz Hazard - Range: > 1.5g	2.10E-07	7.90E-01	1.66E-07
			Total	1.01E-06

To adjust the 1 Hz:
ratio = 0.136/0.11
A_{m1} = 0.635*0.136/0.11 = 0.785
$HCLPF_1$ = 0.309

Table 2-4 Convolution of the WF3 Plant HCLPF over the Range of the 1 Hz Hazard Curve

Scenario	Description	Earthquake Frequency (/yr)	CCDP	SCDF (/yr)
%G01	%G01 - Hazard Curve: W3 1 Hz Hazard - SA Range: 0.1g to 0.3g	1.68E-04	7.92E-05	1.33E-08
%G02	%G02 - Hazard Curve: W3 1 Hz Hazard - SA Range: 0.3g to 0.5g	5.24E-06	3.87E-02	2.03E-07
%G03	%G03 - Hazard Curve: W3 1 Hz Hazard - SA Range: 0.5g to 0.7g	8.87E-07	2.40E-01	2.13E-07
%G04	%G04 - Hazard Curve: W3 1 Hz Hazard - SA Range: 0.7g to 0.9g	2.92E-07	5.11E-01	1.49E-07
%G05	%G05 - Hazard Curve: W3 1 Hz Hazard - SA Range: 0.9g to 1.1g	1.25E-07	7.23E-01	9.01E-08
%G06	%G06 - Hazard Curve: W3 1 Hz Hazard - SA	6.63E-08	8.54E-01	5.66E-08

	Range: 1.1g to 1.3g			
%G07	%G07 - Hazard Curve: W3 1 Hz Hazard - SA Range: 1.3g to 1.5g	3.72E-08	9.25E-01	3.44E-08
%G08	%G08 - Hazard Curve: W3 1 Hz Hazard - SA Range: > 1.5g	8.26E-08	9.68E-01	8.00E-08
			Total	**8.39E-07**

To adjust the 10 Hz:
ratio = 0.175/0.11
A_{m10} = 0.635*0.175/0.11 = 1.01
$HCLPF_{10}$ = 0.397

Table 2-5 Convolution of the WF3 Plant HCLPF over the Range of the 10 Hz Hazard Curve

Scenario	Description	Earthquake Frequency (/yr)	CCDP	SCDF (/yr)
%G01	%G01 - Hazard Curve: W3 10 Hz Hazard - SA Range: 0.1g to 0.3g	1.43E-04	5.23E-06	7.50E-10
%G02	%G02 - Hazard Curve: W3 10 Hz Hazard - SA Range: 0.3g to 0.5g	1.10E-05	8.29E-03	9.15E-08
%G03	%G03 - Hazard Curve: W3 10 Hz Hazard - SA Range: 0.5g to 0.7g	2.72E-06	9.06E-02	2.46E-07
%G04	%G04 - Hazard Curve: W3 10 Hz Hazard - SA Range: 0.7g to 0.9g	9.58E-07	2.73E-01	2.62E-07
%G05	%G05 - Hazard Curve: W3 10 Hz Hazard - SA Range: 0.9g to 1.1g	3.95E-07	4.85E-01	1.92E-07
%G06	%G06 - Hazard Curve: W3 10 Hz Hazard - SA Range: 1.1g to 1.3g	1.96E-07	6.64E-01	1.30E-07
%G07	%G07 - Hazard Curve: W3 10 Hz Hazard - SA Range: 1.3g to 1.5g	1.02E-07	7.91E-01	8.10E-08
%G08	%G08 - Hazard Curve: W3 10 Hz Hazard - SA Range: > 1.5g	1.89E-07	8.90E-01	1.68E-07
			Total	**1.17E-06**

Based on published bounding SCDF estimates (Table 2-1) and the SCDF estimate calculated using the updated 2014 W3 seismic hazard (Tables 2-2 through 2-5), seismic risk cannot be excluded from the calculation of configuration-specific risk. An estimated seismic analysis is therefore developed.

3. Seismic Estimate Analysis

If an external hazard does not screen as risk-insignificant, RICT calculations may use conservative or bounding analyses (Ref. 6). An analysis of the external event contribution to configuration risk is performed and these results are incorporated into the RMTS program. This may be accomplished via performing a reasonable conservative analysis and applying it along with the internal events risk contribution in calculating the configuration risk and the associated RICT.

3.1 Estimate SCDF

Based on the review of seismic risk estimates discussed in Section 2, a reasonable estimate of WF3 SCDF is 1.26E-06/yr. The 2.9E-05/yr estimate from the 1994 data is overly conservative. The WF3 NFPA 805 Safety Evaluation applies the 2008 USGS value of 2.0E-05 which is also very conservative. The 1989 data is dated and therefore is eliminated in favor of more recent evaluations. Being the most up to date data, the maximum of the four estimates based on the 2014 data was chosen. The maximum was chosen over the IPEEE weighted average used in the previous revision as being more conservative. See Tables 2-2 through 2-5 for the results of these estimates.

The FRANX evaluations for the 5 Hz and 10 Hz cases have a CCDP of significantly less than one for the >1.5g hazard interval. Even if the CCDP is assumed to be 1.0 for these hazard intervals, the results for the 5 and 10 Hz cases do not change enough to alter the conclusion. The PGA case remains the highest estimate.

There are several sources of conservatism inherent in this approach.

- The seismic contribution to delta risk for the RICT calculation for any configuration is taken as the full seismic plant CDF / LERF. That is, for the purpose of any RICT calculation, delta seismic CDF is assumed to be equal to the estimated seismic CDF from the convolution of the seismic hazard curves with the limiting HCLPF.
- The full annual seismic frequency is applied to the seismic contribution for all RICT calculations, regardless of the duration of the RICT.
- The presumption of the plant HCLPF leading directly to seismic core damage means that no other failures, whether related to components in an LCO or not, and whether treated as correlated or not, are required for the calculation of seismic CDF (i.e., the use of plant level HCLPF assumes that enough equipment failures occur to lead directly to core damage.), and these would not increase the estimated seismic CDF.

The estimate chosen for RICT application is the maximum hazard with a CDF of 1.26E-06/yr.

3.1.1 SCDF Uncertainty

The uncertainty associated with seismic risk is typically dominated by the uncertainty in the initiating event frequency, local building response, and component seismic capacity. Using the method of estimated plant fragility in place of building response and component capacity, the uncertainty of these parameters cannot be easily determined. Without further work it is difficult to ascertain exactly how much confidence is embedded in the estimate. This method is intended to be a conservative estimate of the overall SCDF/SLERF.

3.2 Qualitatively Evaluate the SLERF Contribution

The W3 internal events CDF is 3.03E-06/yr and internal events LERF is 3.03E-08/yr, two orders of magnitude lower. The potential exists for a seismic event to contribute to a bypass of containment, such as impacts to containment isolation valve(s) or spurious operations produced from contact chatter. A conditional large early release probability (SCLERP) of 0.1 is judged to be a reasonably conservative estimate for W3 and conforms to the NFPA 805 License Amendment (Ref. 13). However, Appendix A estimates SLERF as 1.8E-07/yr using a convolution of containment failure and containment bypass across the PGA Hazard curve. Additional contributions to SLERF could come from Station Blackout and Anticipated Transient

Without Scram. The total contributions result in a SCLERP ratio to SCDF of 0.142 per the calculations in Appendix A, section 2.1.4.2. A range of SCLERP estimate of 0.1 to 0.15 is given in Appendix A.

Based on the estimate being slightly less than the upper end of the 0.1 – 0.15 range, the upper end of the range is selected as the conservative estimate. This results in a SLERF estimate of 1.9E-07/yr.

SCDF * 0.15 = SLERF   1.26E-06 * 0.15 = 1.89E-07 ~ 1.9E-07

4. Conclusion

Based on NEI-06-09 guidelines, TSTF-505 LAR submissions are to justify excluding any risk sources determined to be insignificant to the calculation of configuration-specific risk, and also to provide a discussion of any conservative or bounding analyses to be applied to the calculation of risk-informed completion times (RICTs) and risk management action times (RMATs) for sources of risk not addressed by the PRA models.

Based on published bounding SCDF estimates and the SCDF estimate calculated using the updated 2014 W3 seismic hazard, seismic risk cannot be excluded from the calculation of configuration-specific risk. A conservative analysis was therefore developed, which produced an SCDF and SLERF of 1.26E-06/yr and 1.9E-07/yr , respectively.

Seismic risk will be included in RICT and RMAT calculations by adding an incremental 1.26E-06/year and 1.9E-07/year seismic contribution to the configuration-specific delta CDF/delta LERF, respectively, attributed to internal and fire events contributions. This method ensures that an incremental seismic CDF/LERF equal to the estimated SCDF/SLERF is added to internal and fire events incremental CDF/LERF contribution for every RICT occurrence. The total configuration-specific delta CDF/LERF attributed to internal, fire and the seismic CDF/LERF values are compared against the ICDP/ILERP acceptance criteria of 1E-05/1E-06.

A permanent SCDF of 1.26E-06/yr and SLERF of 1.9E-07/yr will also be added to the WF3 configuration risk management model to quantify instantaneous CDF/LERF whenever a RICT is in effect.

5. References

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4. NUREG-1488, 1994, "Revised Livermore Seismic Hazard Estimates for Sixty-Nine Nuclear Power Plant Sites East of the Rocky Mountains", U.S. Nuclear Regulatory Commission, Washington, D.C.
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8. NUREG-1407, "Procedural and Submittal Guidance for the Individual Plant Examination of External Events (IPEEE) for Severe Accident Vulnerabilities", June 1991.
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10. EPRI FRANX Software, Rev. 4.4, 2017.
11. EPRI Letter Fleet Seismic Core Damage Frequency Estimates for Central and Eastern U.S. Nuclear Power Plants Using New Site-Specific Seismic Hazard Estimates, March 11, 2014 (ML14083A586).
12. ASME/ANS RA-Sa–2009, Standard for Level 1/Large Early Release Frequency Probabilistic Risk Assessment for Nuclear Power Plant Applications, Non-Mandatory Appendix 10-B
13. Amendment 248 to Waterford Steam Electric Station, Unit 3 Facility Operating License, NPF-38

Appendix A

WF3 50.69 RAI APLC-03 RESPONSE

REPORT

WF3 50.69 RAI APLC-03 RESPONSE INFORMATION

PREPARED FOR

Entergy Operations
Waterford Unit 3

Project #: 1AJS21123
Report #: 021123-RPT-01 Rev. 1
Date: August 16, 2021

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Reviewer for Remainder:	Larry Lee	Digitally signed by Larry Lee Date: 2021.08.16 11:17:44-07'00'
Review Method	Design Review ☐	Alternate Calculation ☒
Approver:	N/A	

Revision Record Summary

Revision	*Revision Summary*
0	Initial development and issue of report to be used as information to prepare the official response to NRC for Waterford Unit 3 50.69 LAR RAI APLC-03
1	Update in response to Entergy and Westinghouse comments

Table of Contents

1.0 Purpose

The purpose of this report is to provide supporting analyses and documentation for use in developing a response to the NRC RAI APLC-03 (Reference 3) in response to the Entergy Waterford Unit 3 (WF3) 10 CFR 50.69 license amendment request (LAR) application (Reference 1). RAI APLC-03 requests additional information regarding seismic risk estimation for WF3.

The discussions in this report are supported by spreadsheet calculations (Reference 49) and Waterford Unit 3 "WF3Rev6_R0" probabilistic risk assessment (PRA) sensitivity quantifications (Reference 50).

2.0 Response to NRC RAI APPLC-03

In addition to other RAIs, the NRC issued the following RAI APLC-03 (Reference 3) following the June 2021 audit of the WF3 50.69 LAR:

> Section 50.69(b)(2)(ii) of 10 CFR requires that a LAR include a "description of the measures taken to assure that the quality and level of detail of the systematic processes that evaluate the plant for internal and external events during normal operation, low power, and shutdown … are adequate for the categorization of SSCs." Section 50.69(b)(2)(iv) of 10 CFR requires that a LAR include a "description of, and basis for acceptability of, the evaluations to be conducted to satisfy 10 CFR 50.69(c)(1)(iv)." The Statement of Consideration (SOC) on 10 CFR 50.69(b)(2)(iv) of the Final rule published in the Federal Register on November 22, 2004 (69 FR 68008)) states that the licensee, "is required to include information about the evaluations they intend to conduct to provide reasonable confidence that the potential increase in risk would be small." The SOC further clarifies that a "licensee must provide sufficient information to the NRC, describing the risk sensitivity study and other evaluations and the basis for their acceptability as appropriately representing the potential increase in risk from implementation of the requirements in the rule."
>
> In Section 3.2.3, "Seismic Hazards," of the LAR Enclosure, the licensee states, in part, that "low seismic CDF and LERF estimates lead to reasonable confidence that seismic risk contributions would allow reducing an HSS to LSS due to the 10 CFR 50.69 Integral Assessment if the equipment is HSS only due to seismic considerations." Section 2.2.2 of EPRI Report 3002017583 identifies the contribution of seismic to total plant risk as a basis for the use of the proposed alternative seismic approach for Tier 1 sites. However, the NRC staff notes that the LAR does not provide information to show that the plant-specific seismic risk constitutes a small fraction of the total plant risk and thus that the proposed alternative seismic approach is applicable to Waterford 3.
>
> In Section 3.2.3 of the LAR Enclosure, the licensee further states that "Waterford 3 completed a bounding seismic risk evaluation to support development of a Risk-Informed Completion Time (TSTF-505) license amendment request and program." Based on the Technical Specifications Task Force (TSTF) Traveler TSTF-505 LAR (ADAMS Accession No. ML21039A648) for Waterford 3, it appears that the seismic penalty was based on a plant's high-confidence of low-probability of failure (HCLPF) capacity of 0.25g, as opposed to 0.1g in the Generic Issue 199 report (Safety/Risk Assessment Results for Generic Issue 199, "Implications of Updated Probabilistic Seismic Hazard Estimates in Central and Eastern United States on Existing Plants," dated September 2, 2010 (ADAMS Accession No. ML100270582)).
>
> a. Provide justification for a plant HCLPF capacity of 0.25g used for Waterford 3.
>
> b. Justify that the plant-specific seismic risk is low relative to the overall plant risk such that the categorization results will not be significantly impacted to support the applicability of the proposed alternative seismic approach to Waterford 3.

The responses to the above two questions are provided below. The official response to the NRC may not need to incorporate all the following details.

2.1 APLC-03 PART (A) RESPONSE

The NRC is correct that the WF3 TSTF-505 LAR (Reference 2), referenced in the WF3 50.69 LAR (Reference 1), inappropriately cited 0.25g PGA as the WF3 plant level fragility HCLPF value cited by the NRC in the Generic Issue 199 risk assessment report (Reference 7). In addition, the WF3 50.69 LAR and WF3 TSTF-505 LAR inappropriately characterized the RICT seismic penalty estimates as "bounding". The following response to APLC-03 part (a) addresses these two points in the following manner:

+ Justification of reasonableness of 0.25g WF3 plant level fragility HCLPF
  - General aspects of SMA-based plant level seismic fragilities (Section 2.1.1)
  - Range of plant level fragilities from NTTF 2.1 SPRAs (Section 2.1.2)
  - Review of margin in WF3 plant beyond the design basis SSE (Section 2.1.3)
+ Seismic penalty estimates not bounding (Section 2.1.4)

2.1.1 General Aspects of SMA-Based Plant Level Seismic Fragilities

From the perspective of seismic margin assessment (SMA) bases, this sub-section explains that the 0.1g PGA SSE-based review level earthquake (RLE) assigned for the WF3 IPEEE seismic analysis by the NRC (Reference 10) is an overly conservative estimate to use as the WF3 plant level seismic fragility HCLPF.

For nuclear power plants that do not have a current or maintained seismic PRA, it has become common in U.S. nuclear power risk assessment and management to approximate the plant seismic core damage frequency (SCDF) by convolving an estimate of the plant level seismic HCLPF (typically from an existing SMA) with the plant seismic hazard curve. Convolution is a mathematical term that refers to combining (e.g., multiplying) two or more inter-related functions. In the case of this SCDF approximation approach, the inter-related functions are the seismic hazard curve and the plant level seismic fragility. The hazard curve is a function of increasing magnitude of the hazard load with corresponding reduction in occurrence frequency. The plant level seismic fragility function is an increasing probability of failure with increasing magnitude of the hazard load.

The plant level fragility convolution approach is used in the WF3 LAR, as well as many other TSTF-505 LARs. This is also the process used by the NRC in the Generic Issue 199 risk assessment report (Reference 7) to estimate SCDFs for the various U.S. plants.

WF3 performed a seismic margins assessment (following EPRI NP-6041 guidance, Reference 6) for the IPEEE program and per the NRC request (Reference 10) used the WF3 design basis safe shutdown earthquake (SSE) as the seismic margins earthquake (SME). In this situation, the SMA is effectively an investigation and confirmation of the design basis and not an analysis of the inherent margin in the plant beyond the design basis.

The purpose of an SMA is to investigate the margin beyond the design basis, as described in the introduction of EPRI NP-6041:

> *"It is a widely held belief in the technical community that current engineering methods and design requirements are substantially conservative and that the resulting seismic margin can accommodate ground motions well above safe shutdown earthquake (SSE) levels. ... The seismic margin earthquake (SME) is chosen to be sufficiently larger than the SSE to establish a significant seismic margin. Methodology procedures determine the weakest link components and establish the HCLPF level of ground motion for which the plant can safely shut down."*

In addition, Part 10 (Seismic Margin Assessment Requirements At-Power) of the ASME/ANS PRA Standard (Reference 11) includes an explicit requirement (i.e., Supporting Requirement SM-A1) that the review level earthquake selected for an SMA be larger than the plant SSE:

SM-A	*Requirement*
SM-A1	SELECT a review level earthquake as an earthquake larger than the safe shutdown earthquake for the plant.

The use of the WF3 SSE as the SME in the WF3 IPEEE SMA was an evaluation constraint of the IPEEE program and the result of the WF3 IPEEE SMA should not be viewed as a reasonable estimate of the WF3 plant level fragility HCLPF given the very low SME (i.e., the SSE) used. This fact is also recognized in the NRC summary of the IPEEE program results, NUREG-1742 (Reference 8):

"It should also be noted that if a plant performed a reduced-scope evaluation, this evaluation was performed using input from the plant's seismic design basis (SSE spectra). Therefore, a reduced-scope evaluation does not convey the degree of seismic margin (i.e., does not produce a plant HCLPF)."

2.1.2 Range of Plant Level HCLPFs from NTTF 2.1 SPRAs

The previous sub-section discusses that the WF3 IPEEE seismic margin earthquake of 0.1g PGA was not intended to be interpreted as a reasonable estimate of the WF3 plant level seismic fragility HCLPF given that the SSE was used as the SME. This section discusses a review of plant level seismic HCLPFs from recent SPRAs to show that plant level seismic HCLPFs are typically in the range of 0.2g to 0.3g PGA (or higher) and that a 0.25g PGA HCLPF value is reasonable for WF3. An overview of the HCLPF concept is provided first.

<u>Overview of HCLPF Concept</u>

Seismic fragility of a structure or equipment item is defined as the conditional probability of its failure at a given value of the seismic input or response parameter (e.g., PGA, stress, moment, or spectral acceleration). In SPRAs and SMAs the fragilities are often presented in terms of g PGA at the ground. Because there are many variables involved in the estimation of a fragility, an SSC fragility is described by a family of fragility curves for different confidence levels. This family of fragility curves may be described by three parameters: the median acceleration capacity Am (or the HCLPF may be used), and logarithmic standard deviations, ßr and ßu, for randomness and uncertainty, respectively. The full form of the fragility equation used in most nuclear power plant (NPP) seismic probabilistic analyses is shown below (Eq. 4-3 of Reference 15):

$$f' = \Phi\left[\frac{\ell n\left(\frac{a}{A_m}\right) + \beta_U \Phi^{-1}(Q)}{\beta_R}\right] \qquad \text{(Eq. 1)}$$

where:

a = response acceleration in question (in units of g)

Am = median acceleration, i.e., response acceleration equal to the 50% failure probability (in units of g)

ßr = logarithmic standard deviation representing randomness

ßu = logarithmic standard deviation representing uncertainty

Q = P[f < f′ | a]; i.e., the subjective probability (confidence) that the conditional probability of failure, f, is less than f′ for acceleration a.

Φ^{-1}[.]= the inverse of the standard Gaussian cumulative distribution of the term in brackets.

The version of the above equation describing specifically the Mean confidence level curve is (refer to Section 4.1.1 of Reference 15):

Fragility (i.e., failure probability) = Φ [ln(A/Am)/ßc] (Eq. 2)

Where, the composite beta, ßc = (ßu^2 + ßr^2)^0.5

HCLPF and Am are related as follows (Eqs. 4-12 and 4-13 of Reference 15):

Am =HCLPF / (exp -1.65(ßr + ßu)) (Eq. 3)

- or -

HCLPF / (exp -2.33ßc) (Eq. 4)

Refer to Reference 15, as necessary, for further discussion of the fragility mathematical model. For illustration, an example (not WF3 specific) family of fragility curves is shown here in Figure 2-1 (Figure 4-1 of Reference 15). The high confidence low probability of failure (HCLPF) point on a family of fragility curves is intended to represent an earthquake level at which there is a 95% confidence of less than 5% failure probability. On the example fragility curve in Figure 2-1 the HCLPF point can be seen as a value of 0.32g PGA with a 0.05 failure probability on the 95% confidence level curve. The HCLPF on the Mean confidence level curve represents a 1% (0.01) failure probability (note where the “HCLPF 0.32g” arrow in Figure 2-1 intercepts the dotted Mean fragility curve).

In SMA analyses the resulting plant level HCLPF is either: 1) the SME (if all SSCs in the SMA analysis are determined to meet the SME) or 2) the lowest HCLPF of the analyzed SSCs (or lowest SSC HCLPF using the “min-max” method of the SMA analyzed success paths; refer to Reference 12). In the case of the WF3 IPEEE SMA, the analysis concluded that the SSE SME was met (i.e., no outliers below the SME).

<u>Range of Plant Level HCLPFs from NTTF 2.1 SPRAs</u>

As WF3 does not have an SPRA or an SMA sufficient to provide a reasonable plant level seismic fragility, a review of recent SPRAs submitted to the NRC in response to NTTF 2.1 seismic request (References 17 thru 27) was performed as part of this RAI response to show that plant level seismic HCLPFs are typically in the range of 0.2g to 0.3g PGA (or higher) and that the 0.25g PGA HCLPF value used in the WF3 TSTF-505 seismic penalty calculation is reasonable. This review focuses on the NTTF 2.1 SPRAs from sites in the central and eastern U.S. (CEUS), the geographical region in which WF3 is located, and did not include the western U.S. plants which have higher plant level HCLPFs in general due to their location in high seismicity zones and their associated seismic designs (e.g., the Diablo Canyon plant level fragility is >1g PGA, Reference 31).

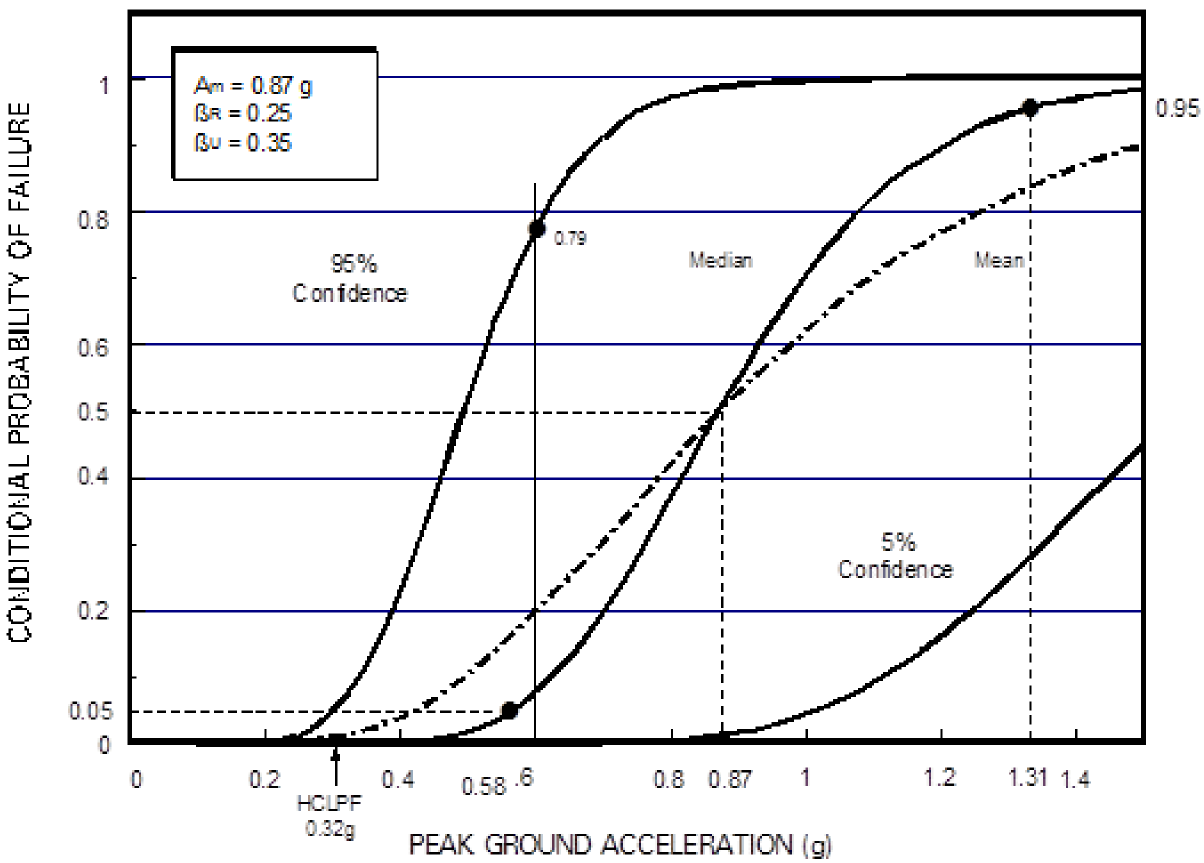


Figure 2-1 - Example Family of Fragility Curves (Reference 15)

As discussed above, the HCLPF point on the fragility Mean confidence level curve is the 1% (0.01) probability of failure point. An SPRA is intended as a Mean based analysis and thus the plant level seismic fragility from an SPRA is the g level of the site seismic hazard curve that produces a 0.01 conditional core damage probability (CCDP). This is illustrated in Figure 2-2. As can be seen from Figure 2-2, the plant HCLPF can be determined by either 1) identifying the 1E-2 CCDP point on the seismic hazard curve, or 2) alternatively, identifying the 0.50 CCDP point on the seismic hazard curve and back-calculating the HCLPF using Eq. 4 and assuming a ßc value (ßc =0.4 is commonly used in such approximations, e.g., refer to Reference 7). This review uses the first approach.

The NTTF 2.1 SPRA information was obtained by accessing the submittal reports from the www.nrc.gov website and reviewing the information contained in the submittals. It was necessary in some cases to contact PRA personnel associated with the analyses to clarify aspects. The SPRA risk quantification results information of interest in this review are conditional core damage probabilities for various points on the hazard exceedance frequency curve (all the submittals reviewed used a PGA based hazard curve).

All of the studies employed the typical SPRA process of dividing the hazard curve into numerous discrete seismic magnitude range intervals with associated seismic initiator occurrence frequency variables (e.g., %G1, 0.05g to 0.2g PGA; %G2, 0.2g to 0.3g PGA; %G3, 0.3g to 0.4g PGA, etc.) and assigned representative magnitudes to use for fragility calculations. The number and widths of seismic hazard intervals typically differed for each of the SPRAs reviewed.

Figure 2-2

Example Graphical Illustration of Plant HCLPF Determination from SPRA Results

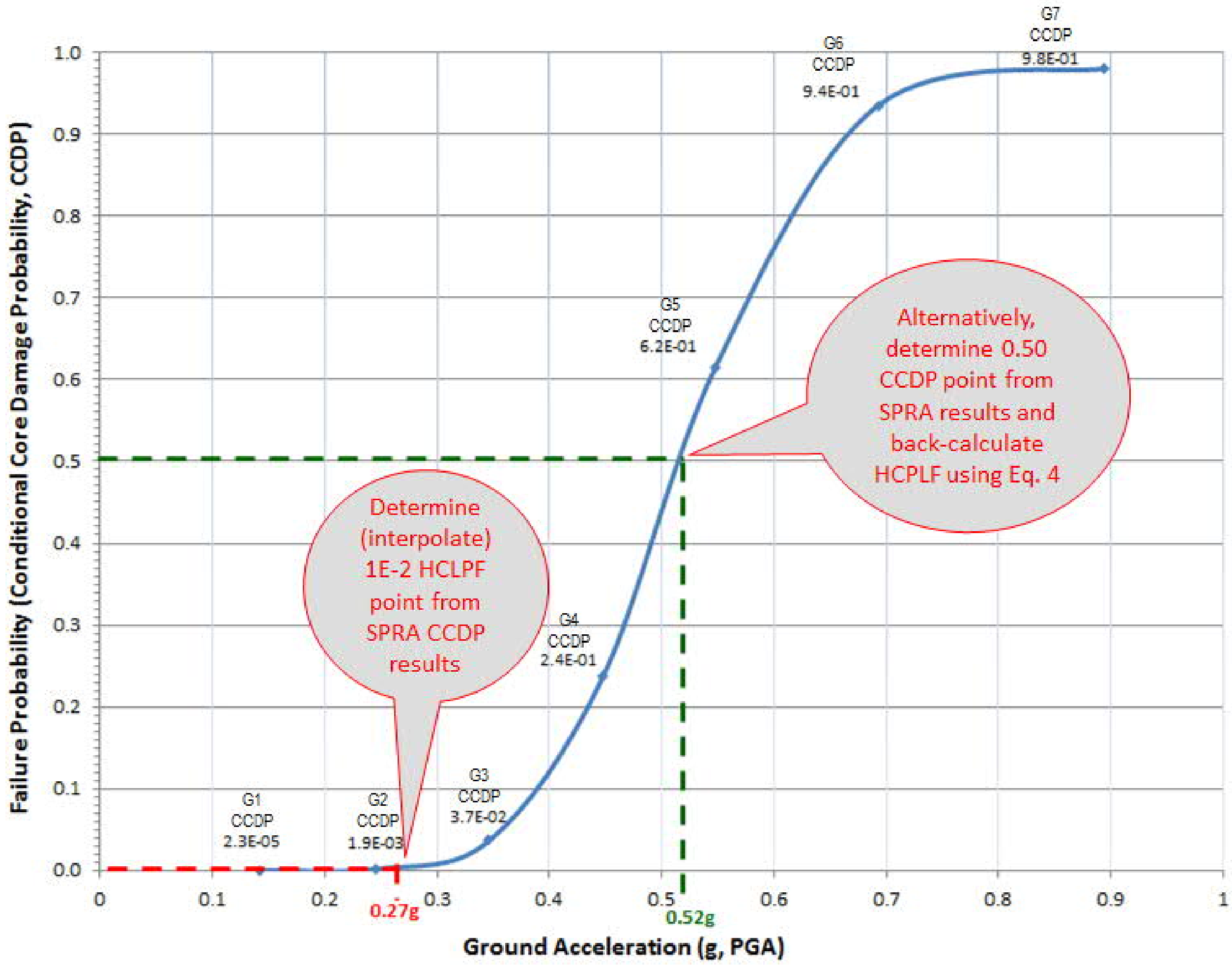


Figure 2-2 - Example Graphical Illustration of Plant HCLPF Determination from SPRA Results

The CCDP of a seismic hazard interval is calculated as the SCDF (/yr) for that hazard interval divided by the initiator frequency (/yr) of the hazard interval (e.g., 7.59E-7/yr / 1.50E-05/yr = 0.051 CCDP). The CCDP per seismic hazard interval information is reported in tables and/or graphs in Section 5 of the NTTF 2.1 SPRA submittal reports. However, in each SPRA, the 1% (0.01) CCDP point had to be linearly interpolated between two hazard intervals. This interpolation is illustrated below for the Sequoyah NTTF 2.1 SPRA, which results in a plant level seismic HCLPF of 0.40g PGA for the Sequoyah plant (Reference 24):

Sequoyah NTTF 2.1 SPRA submittal (Table 5.4-1 of submittal)

Hazard Interval ID	Hazard Interval Representative Magnitude	Hazard Interval Initiator Frequency (/yr)	Hazard Interval SCDF (/yr)	Hazard Interval CCDP
G03	0.387	5.40E-05	3.53E-07	6.54E-03
	0.40	<--- linear interpolated value		1.00E-02
G04	0.592	1.50E-05	7.59E-07	5.06E-02

The results of this review of NTTF 2.1 SPRA plant level HCLPFs are summarized in Table 2-1. As can be seen from Table 2-1, the NTTF 2.1 SPRA plant level HCLPFs are primarily in the 0.2g to 0.3g PGA range, some significantly higher. This conclusion regarding the typical range of plant level HCLPFs of U.S. current generation nuclear plants is echoed in the NRC's summary of the IPEEE program results in NUREG-1742. Figure 2.10 of NUREG-1742, Vol. 1 (Reference 8) summarizes that of the plants who performed an SMA for the IPEEE (not including the reduced-scope assigned plants constrained to an SME equal to the SSE):

+ 92% (33 of 36 SMAs) of the plant level seismic HCLPFs are ≥0.2g PGA
+ 53% (19 of 36 SMAs) are in the 0.2g-0. 29g PGA range and 39% (14 of 36 SMAs) are ≥0.3g PGA.

Table 2-1
Plant Level HCLPFs from NTTF 2.1 SPRAs

Plant[1]	Plant Type	SSE (g, PGA)	Plant Level HCLPF (g, PGA)	Comment
Beaver Valley (Ref. 17)	PWR	0.12 (Ref. 7)	0.29	Table 5-11 of Ref. 17 (linear interpolation between G02 and G03 hazard interval CCDPs and use of 1E-2 HCLPF for Mean SPRA analysis).
Browns Ferry (Ref. 18)	Mk I BWR	0.2 (Ref. 7)	0.25	Table 5.4-1 of Ref. 18 (linear interpolation between G02 and G03 hazard interval CCDPs and use of 1E-2 HCLPF for Mean SPRA analysis).
Callaway (Ref. 19)	PWR	0.2 (Ref. 7)	0.25	Table 5-1 of Ref. 19 (linear interpolation between G02 and G03 hazard interval CCDPs and use of 1E-2 HCLPF for Mean SPRA analysis).
DC Cook (Ref. 20)	PWR	0.2 (Ref. 7)	0.19	Tables 3-4 and 5.4-4 of Ref. 20 (linear interpolation between G01 and G02 hazard interval CCDPs is too coarse of an interpolation given low end of hazard curve and very low CCDP of first interval. Performed sensitivity requantification of DC Cook NTTF 2.1 SPRA G2 interval with revised representative magnitude of 0.19g which produces a 1E-2 CCDP for that interval.
Dresden (Ref. 21)	Mk I BWR	0.2 (Ref. 7)	0.27	Table 5.4-1 of Ref. 21 (linear interpolation between G2 and G3 hazard interval CCDPs and use of 1E-2 HCLPF for Mean SPRA analysis).
North Anna (Ref. 22)	PWR	0.12 (Ref. 7)	0.30	Table 5.4-4 of Ref. 22 - linear interpolation between G01 and G02 hazard interval CCDPs too coarse given very low CCDP of the first interval. Interpolation between G03 and G04 intervals to determine median Am and using Eq. 4 (with assumed Bc=0,4, consistent with Reference 7) results in HCLPF of 0.21. Based on information from North Anna PSA personnel (Reference 28), the NTTF 2.1 SPRA has a number of key conservatisms (not related to hazard occurrence frequencies) that are being addressed and it is predicted overall SCDF will reduce at least 50%. Assuming a 50% SCDF drop would result in a plant HCLPF closer to 0.3g.
Peach Bottom (Ref. 23)	Mk I BWR	0.12 (Ref. 7)	0.21	Table 5.4-1 of Ref. 23 (linear interpolation between G1 and G2 hazard interval CCDPs and use of 1E-2 HCLPF for Mean SPRA analysis).

Table 2-1
Plant Level HCLPFs from NTTF 2.1 SPRAs

Plant[1]	Plant Type	SSE (g, PGA)	Plant Level HCLPF (g, PGA)	Comment
Sequoyah (Ref. 24)	Ice Condenser PWR	0.18 (Ref. 7)	0.40	Table 5.4-1 of Ref. 24 (linear interpolation between G03 and G04 hazard interval CCDPs and use of 1E-2 HCLPF for Mean SPRA analysis).
VC Summer (Ref. 25)	PWR	0.15 (Ref. 7)	0.17	Table 5.4-1 of Ref. 25 (linear interpolation between GA and GB hazard interval CCDPs and use of 1E-2 HCLPF for Mean SPRA analysis).
Vogtle 1,2 (Ref. 26)	PWR	0.2 (Ref. 7)	0.64	Table 5.4-3 of Ref. 26 (linear interpolation between G05 and G06 hazard interval CCDPs and use of 1E-2 HCLPF for Mean SPRA analysis).
Watts Bar (Ref. 27)	Ice Condenser PWR	0.18 (Ref. 7)	0.73	Table 5.4-1 of Ref. 27 (linear interpolation between G4 and G5 hazard interval CCDPs and use of 1E-2 HCLPF for Mean SPRA analysis).

Notes to Table 2-1:

(1) All the NTTF 2.1 SPRA submittals are not included in this summary table; the Oconee, Robinson, Columbia Generating Station and Diablo Canyon submittals are not included here.. The NTTF 2.1 SPRA submittals for the Oconee and Robinson plants are not publicly available. The U.S. West coast plants (Columbia Generating Station and Diablo Canyon) are in high seismicity zones and their plant level seismic HCLPFs are potentially significantly higher than CEUS plants and not directly applicable to the WF3 site.

The results of the one-time IPEEE risk assessments and the GI-199 seismic risk estimates based on the IPEEE analyses (Reference 7) should not be interpreted as definitive estimates of plant-specific seismic risk. These were programmatic studies to highlight seismic risk aspects per NRC Generic Letter 88-20, Supplement 4 (Reference 10) and to fulfill screening steps of the NRC Generic Issues program. It is not uncommon that plant level seismic fragilities from the IPEEE studies and cited in the GI-199 risk estimation report (Reference 7) have been revised by organizations such as EPRI (refer to Reference 16) or utilities such as Entergy (refer to References 32 and 33). In Reference 32 Entergy docketed a reassessment of the IPEEE-based plant level seismic fragility for the Pilgrim plant and increased the IPEEE-based HCLPF from 0.25g PGA to 0.33g PGA. In Reference 33 Entergy docketed a reassessment of the IPEEE-based plant level seismic fragility for the Indian Point 3 plant and increased the IPEEE-based HCLPF from 0.15g PGA to 0.50g PGA.

Based on the preceding discussion, a 0.25g PGA plant level HCLPF is reasonable for the WF3 site and appropriate for use in a seismic penalty SCDF convolution calculation. The following sub-section provides WF3 plant-specific design margin discussions to reinforce the reasonableness of the 0.25g PGA plant level HCLPF value for WF3.

2.1.3 Margin Beyond WF3 Design Basis SSE

To further support the assertion that 0.25g is reasonable for use as a plant level HCLPF for WF3, a review was performed considering the WF3 seismic design basis along with additional seismic evaluation work that has been performed at WF3 to identify likely sources of seismic margin. The review considers seismic walkdown efforts, WF3 seismic demand, and seismic capacity considerations including the use of SMA experience data, as well as seismic interactions, relay chatter, and equipment anchorage.

2.1.3.1 Seismic Walkdowns

Two significant seismic walkdown efforts are considered in this review: the IPEEE walkdowns and the NTTF R2.3 walkdowns.

As discussed above, to meet the requirements of IPEEE per NUREG-1407 [36], WF3 used the reduced-scope seismic margins analysis (SMA) approach. The reduced-scope process focused on seismic walkdowns to identify weak-link items that need strengthening. The process was essentially a confirmation against design-basis with no additional seismic margins work (as previously discussed in Section 2.1.1). The results of the walkdowns are documented in the WF3 IPEEE response letter W3F1-95-0117 [37].

The IPEEE seismic response consisted primarily of walkdowns to confirm design against the plant seismic design basis. Four sets of walkdowns were performed:

1. On-line Walkdown Train B
2. On-line Walkdown Train A
3. Outage Walkdown
4. Structures Walkdown

Walkdown 1 consisted of 388 items and walkdown 2 consisted of 273 items. The seismic walkdowns found WF3 to be seismically rugged and identified no outliers affecting plant operability. A single item could not be screened due to seismic interaction involving a station air pipe close to a switchgear.

Walkdown 3 consisted of 149 items. There were no outliers affecting plant operability found during the walkdown. A number of panels were not able to be screened due to seismic interaction involving personal storage lockers and file cabinets.

Walkdown 4 consisted of 7 structures. The seismic walkdowns and structures review found WF3 to be seismically rugged and identified no outliers affecting plant operability.

Some additional relevant notes from the IPEEE seismic walkdowns:

+ Electrical equipment seismically qualified to IEEE 344-75
+ Walkdowns confirmed that there was no bolt degradation for the tanks and the tanks meet the plant design basis with additional margin.
+ Walkdowns did not identify any issues with distributed systems including piping, cable trays, and HVAC.
+ Seismic interactions with non-seismic equipment are addressed in the WF3 FSAR and block walls were evaluated by WF3 in response to IE Bulletin 80-11.
+ Anchorage was evaluated for a small subset of components. The remaining components were screened as obviously rugged and installed to the design basis. For those that were evaluated, rough demands were used, that were in many cases conservative due the use of 2% damped floor spectra when 5% damped floor spectra were not available, and no concerns were identified.
+ Relay failure and chatter effects were explicitly eliminated from reduced-scope process with the exception of seismic interaction impacts.
+ Soil failure evaluations were explicitly eliminated from reduced-scope process.

In response to the Fukushima Near-Term Task Force Recommendation 2.3 (NTTF R2.3), walkdowns were performed per the guidance document EPRI Report 1025286 [38] during October 2012 and March 2013 as documented in WF3-CS-12-00003 [39].

114 items were evaluated during R2.3 walkdowns and 58 area walk-bys were performed. From these walkdowns and walk-bys, a total of 26 conditions were identified as “potentially adverse seismic conditions”. Each of these were dispositioned through either a licensing basis evaluation or entry in the CAP process depending on their perceived severity. For the 8 conditions that received licensing basis evaluations, all were found to be consistent with licensing basis and required no further action. For the 18 conditions that entered the CAP process, all but 1 were found to have no impact on operability. The single item that required action was a temporary enclosure which had been installed in the Switchgear Room and was unanchored and judged to be a seismic interaction concern; scaffold bars were installed to secure the enclosure to adjacent structure and mitigate the condition.

2.1.3.2 Seismic Demand

There are three primary seismic response spectra (RS) that are considered in this review. The first is the WF3 Safe Shutdown Earthquake (SSE) design spectrum per the WF3 FSAR [45], which represents the minimum seismic level to whish SSCs at WF3 must be designed. The second is the SSE time history (TH) response spectrum. Per the WF3 FSAR: “The time history used to calculate the floor response spectra produces a ground response spectrum which envelopes the design ground response spectra. In order to do this, it has spectral peaks which are substantially higher than the design spectra.” The RS associated with the TH are identified as the WF3 TH SSE RS herein.

The third considered RS is the ground motion response spectrum (GMRS) which was developed through a Probabilistic Seismic Hazard Analysis (PSHA) as documented in the WF3 Seismic Hazard and Screening Report (SHSR) [40] and is taken to represent the most appropriate RS shape associated with seismic risk at WF3. The GMRS as developed for the SHSR is consistent with RLEs used in current SPRAs.

The WF3 SSE, SSE TH RS, and GMRS are plotted in Figure 2-3 with associated PGA values of 0.1g, 0.135g, and 0.11g respectively. The 2 - 8 Hz range is generally considered to be the most damaging for SSCs, and it can be seen that the SSE TH RS exceeds the SSE, which in turn exceeds the GMRS across that frequency range. This implies an intrinsic level of seismic margin built into the design.

A simple seismic fragility considering no factors other than seismic demand allows for the determination of a scaling factor based on the comparison of the demand RS, in this case the GMRS, to the capacity RS from design basis, in this case the SSE TH RS. Considering a frequency range of interest from 1.2 Hz to 10 Hz, which easily encompasses the 2 - 8 Hz range, the demand RS can be scaled by a factor of 1.306 as shown in Figure 2-4, yielding a simple PGA HCLPF of 0.144g (= 0.11g x 1.306).

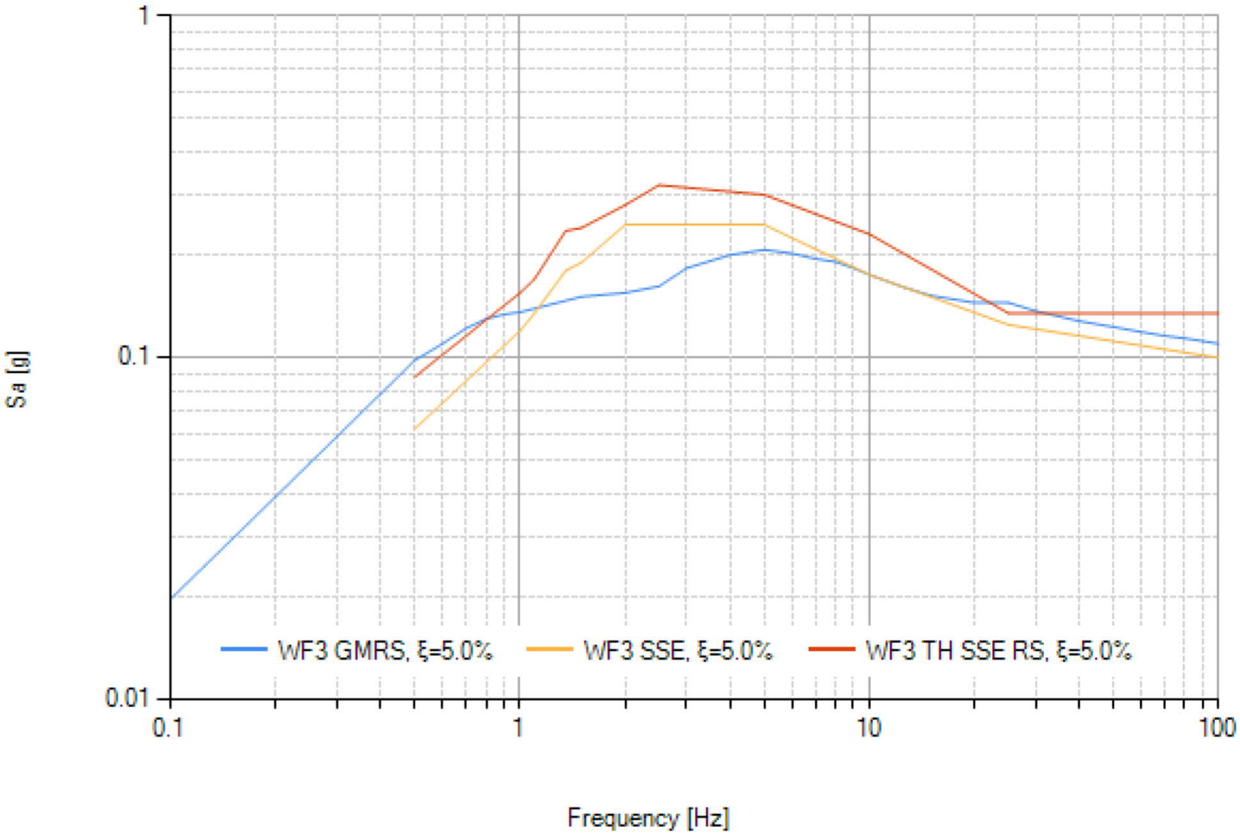


Figure 2-3 - WF3 Seismic Response Spectra

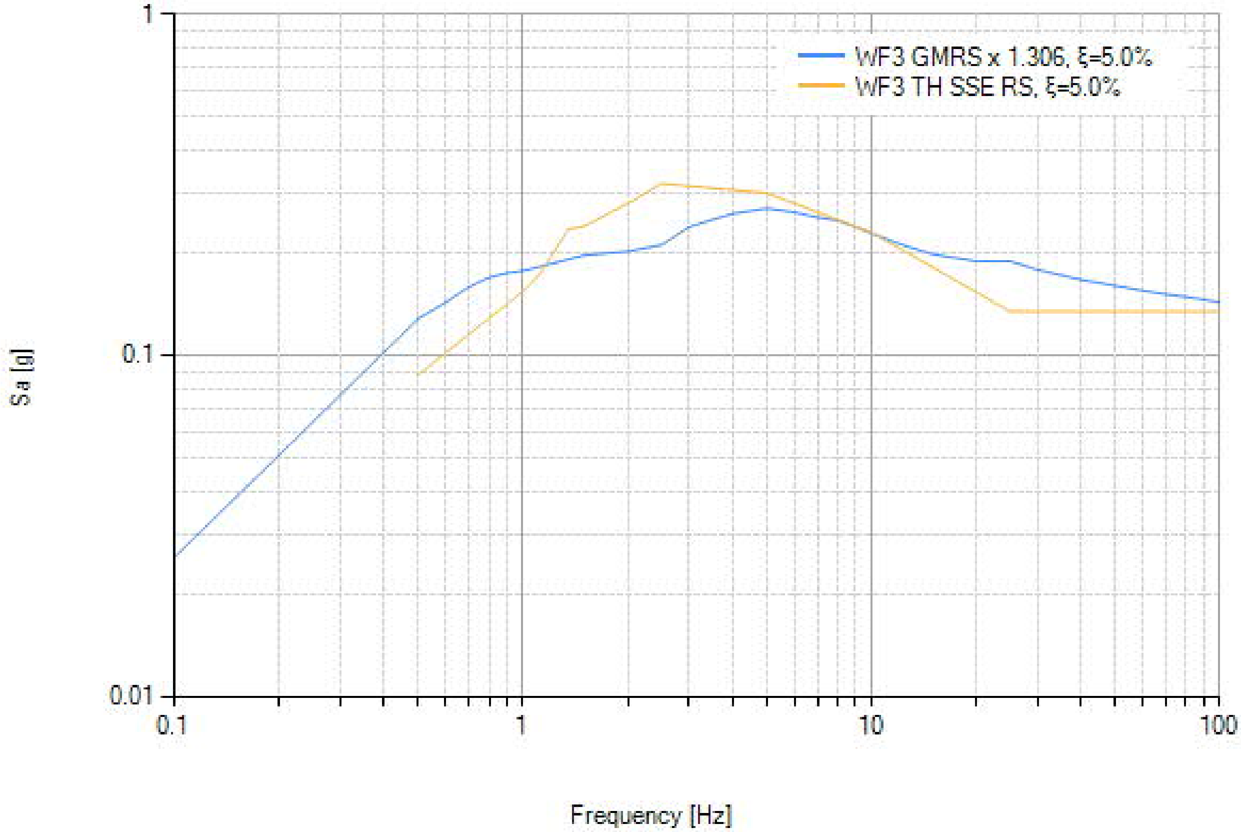


Figure 2-4 - WF3 Scaled GMRS to SSE TH RS Comparison

The seismic design and assessment of SSCs requires the development of floor response spectra, or in-structure response spectra (ISRS), using the ground-level RS as input. Additional seismic margin may exist if conservatism is introduced in the seismic analysis models of structures through stiffness and damping values.

Per the FSAR, with respect to the generation of ISRS: "The building and soil damping values used in the analysis are near the lower bound of the available damping data. The actual values of damping are expected to be much higher than the values used in the analysis." Typical values used in seismic analysis at WF3 are compared to recommended values from US NRC RG 1.61 [41] in Table 3.7-1 from the WF3 FSAR as shown below. It can be seen that the WF3 values are consistently lower than RG 1.61, most notably for structures subjected to SSE loading. Lower damping values will tend to result in higher demands in ISRS, which would yield additional seismic margins in a fragility evaluation as the in-structure demands would be reduced.

TABLE 3.7-1

DAMPIMG VALUES (PERCENT OF CRITICAL DAMPING)

		Operating Basis Earthquake (OBE)		Safe Shutdown Earthquake (SSE)	
		Waterford	RG1.61	Waterford	RG1.61
Welded Steel Plate Assemblies		1	2	1	4
Steel Containment Vessel		2	2	2	4
Welded Steel Frame Structures		2	2	2	4
Bolted or Riveted Steel Framed Structures		2.5	4	2.5	7
Reinforced Concrete Equipment Supports		2	4	5	7
Reinforced Concrete Frames and Buildings		2	4	5	7
Steel Piping	(12" or less)	0.5	1	1.0	2
	(>12")	0.5	2	1.0	3
Soils (Pleistocene Deposit)		7.5		7.5	

Beyond the design-level damping ratio consideration, an appropriate review-level earthquake is expected to be significantly greater than the SSE, as detailed throughout this review, and may justify stiffness and damping values that account for stresses beyond the linear elastic values. Such an adjustment would also be expected to generally reduce the magnitude of floor-level RS.

2.1.3.3 Seismic Capacity

A detailed seismic fragility for a component would be developed by determining capacities for plausible controlling failure modes to determine the seismic level that would cause failure. To perform these calculations for every SSC represents an enormous effort and the SMA process as outlined in EPRI NP-6041-SLR1 (Reference 6) includes a screening process that allows for the assignment of generic seismic capacities for typical components based on experience data for how such components have performed in facilities that experienced real seismic events. Such screening represents the capacity of the component itself and is supplemented by evaluation of seismic interactions, relay chatter, and supports and anchorage.

2.1.3.3.1 Experience-based Seismic Capacities

EPRI NP-6041-SLR1 provides screening tables for civil structures and for equipment and sub-systems. The screening tables have the 3 “screening lanes” and provide spectral accelerations which can be compared to the ground response spectrum and are primarily applicable for structures, which are founded on the ground, and equipment and components located < 40ft above grade. The screening lanes represent spectral capacities of 0.8g, 1.2g, and >1.2g and have different requirements and caveats to be met, with the lowest being the easiest to achieve. The lowest of the screening lanes has a 5% damped peak spectral acceleration (Sa) value of 0.8g. This value can be compared against the peak of the seismic margin earthquake and is most applicable for peaks between 2 and 8 Hz. Original screening tables were based on PGA as opposed to peak spectral acceleration with a PGA of 0.3g being associated with the Sa value of 0.8g.

Based on a review of the screening caveats and FSAR, the major structures at WF3 can be screened in the first screening lane and assigned a spectral capacity of 0.8g. The steel containment could further be screened in the second screening lane with a spectral capacity of 1.2g and PGA of 0.5g if needed. The remainder of the structures cannot be screened to the second screening lanes due to limitations imposed from the structural codes making up the design basis.

Safety-related equipment can also generally be screened in the first screening lane and assigned a spectral capacity of 0.8g in terms of functionality. As Waterford is a plant for which construction began in the mid-1970s, it is expected that good seismic design has been accounted for in the structures, systems, and components beyond just design-basis. This is supported by the reported seismic walkdowns which identified only limited seismic interactions and no general seismic concerns. Based on this, it is reasonable to expect the vast majority of safety-related components to be able to be screened to the first screening lane in a similar manner to the structures.

Figure 2-5 shows the GMRS for WF3 scaled up to meet Sa = 0.8g and PGA = 0.3g. For Sa, the scaling factor on GMRS is 3.865; for PGA, the scaling factor is 2.727. Taking the GMRS as the seismic margin earthquake, the Sa scaling factor can be used to yield simple PGA HCLPF of 0.425g (= 0.11g x 3.865).

A similar comparison can be performed for the SSE and SSE TH to show expected capacity beyond design basis. Figures 2-6 and 2-7 show the SSE and SSE TH RS for WF3 scaled up to meet Sa = 0.8g and PGA = 0.3g. For Sa, the scaling factor on SSE is 3.265 and the scaling factor on SSE TH RS is 2.5; for PGA, the scaling factor on SSE is 3 and the scaling factor on SSE TH RS is 2.222. Taking the SSE as the seismic margin earthquake, the Sa scaling factor can be used to yield a simple PGA HCLPF of 0.326g (= 0.1g x 3.265). Taking the SSE TH RS as the seismic margin earthquake, the Sa scaling factor can be used to yield a simple PGA HCLPF of 0.338g (= 0.135g x 2.5).

As discussed above, anchorage and supports frequently need separate capacity evaluations which will be discussed below.

2.1.3.3.2 Seismic Interactions

Seismic interactions were evaluated during the IPEEE and R2.3 walkdowns and only a small number were identified and were primarily temporary conditions that were easily remedied. There is no indication from the walkdowns that seismic interactions are a general concern and would be identified as controlling fragilities for the majority of plant equipment. To further support this, block walls, which are one of the most significant sources of seismic interaction, were not noted as interactions in either the IPEEE or R2.3 and have been evaluated by WF3 in response to IE Bulletin 80-11. During the 80-11 process, vulnerable walls near safety-related equipment were analyzed and reinforced as required to eliminate them as potential seismic interactions.

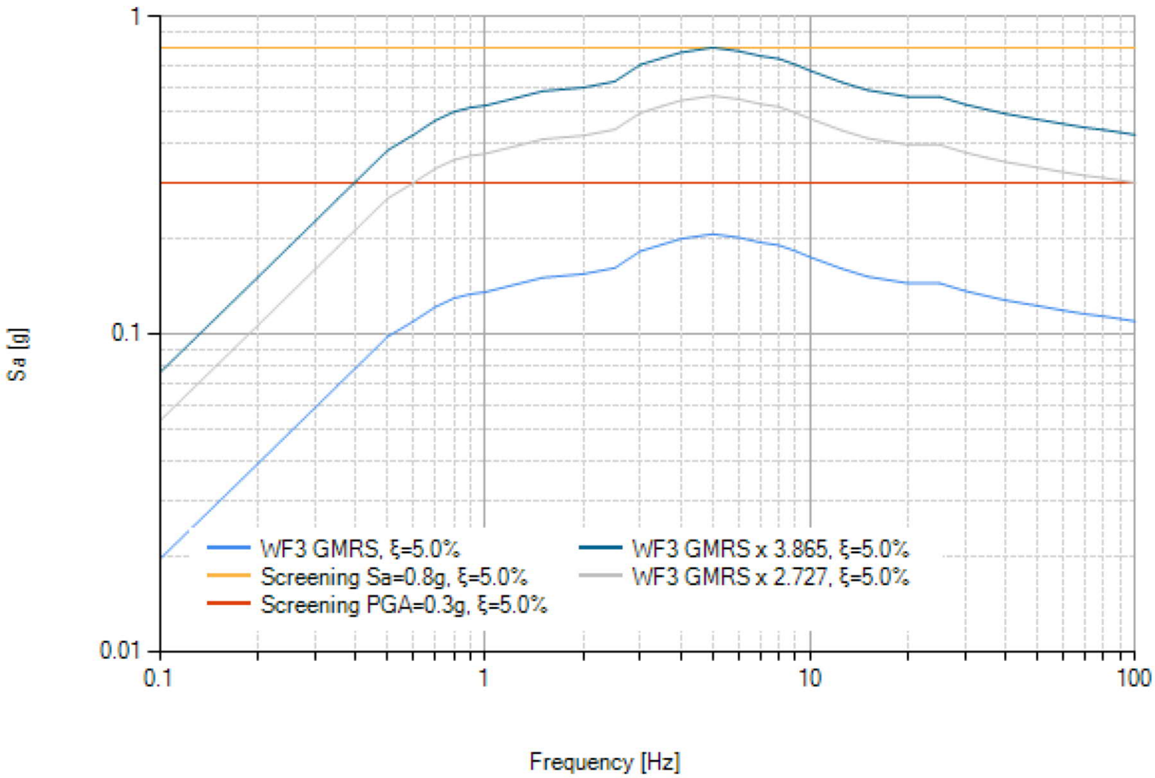


Figure 2-5 - WF3 Scaled GMRS to 0.8g Screening Capacity

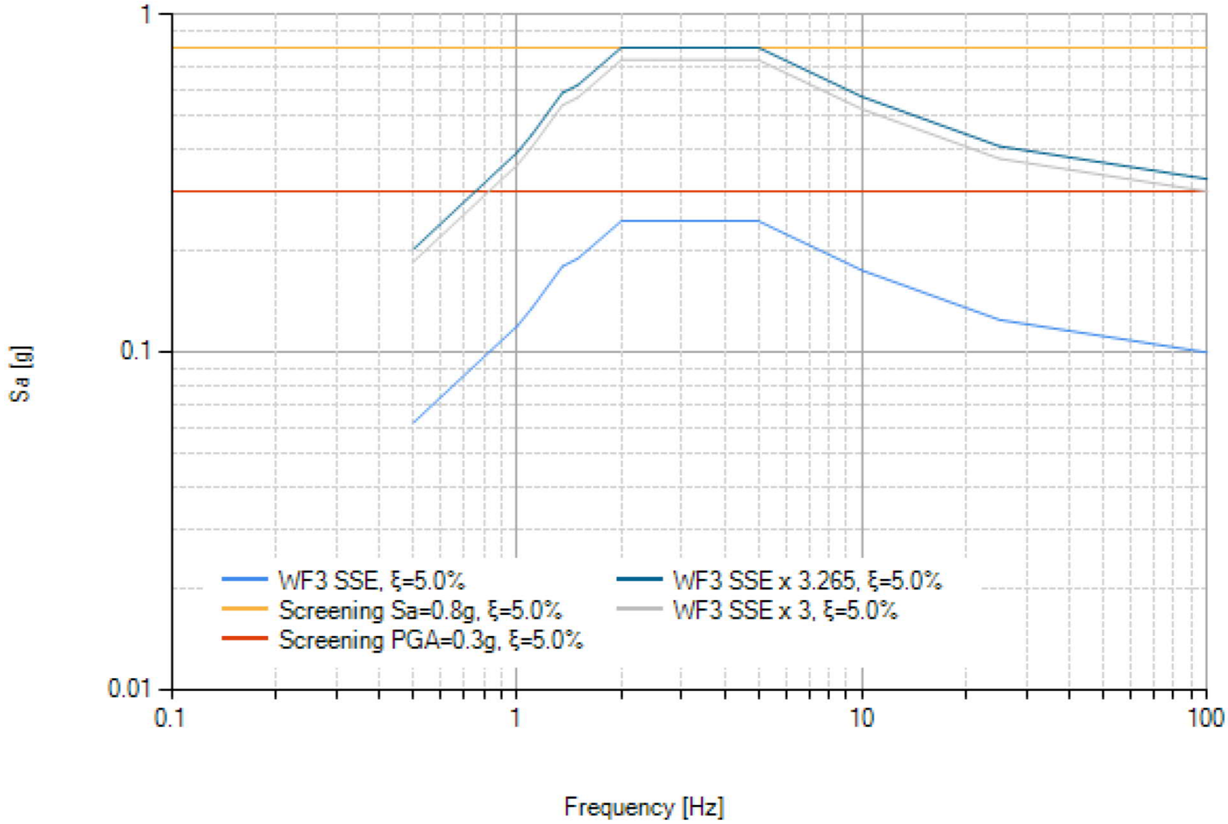


Figure 2-6 - WF3 Scaled SSE to 0.8g Screening Capacity

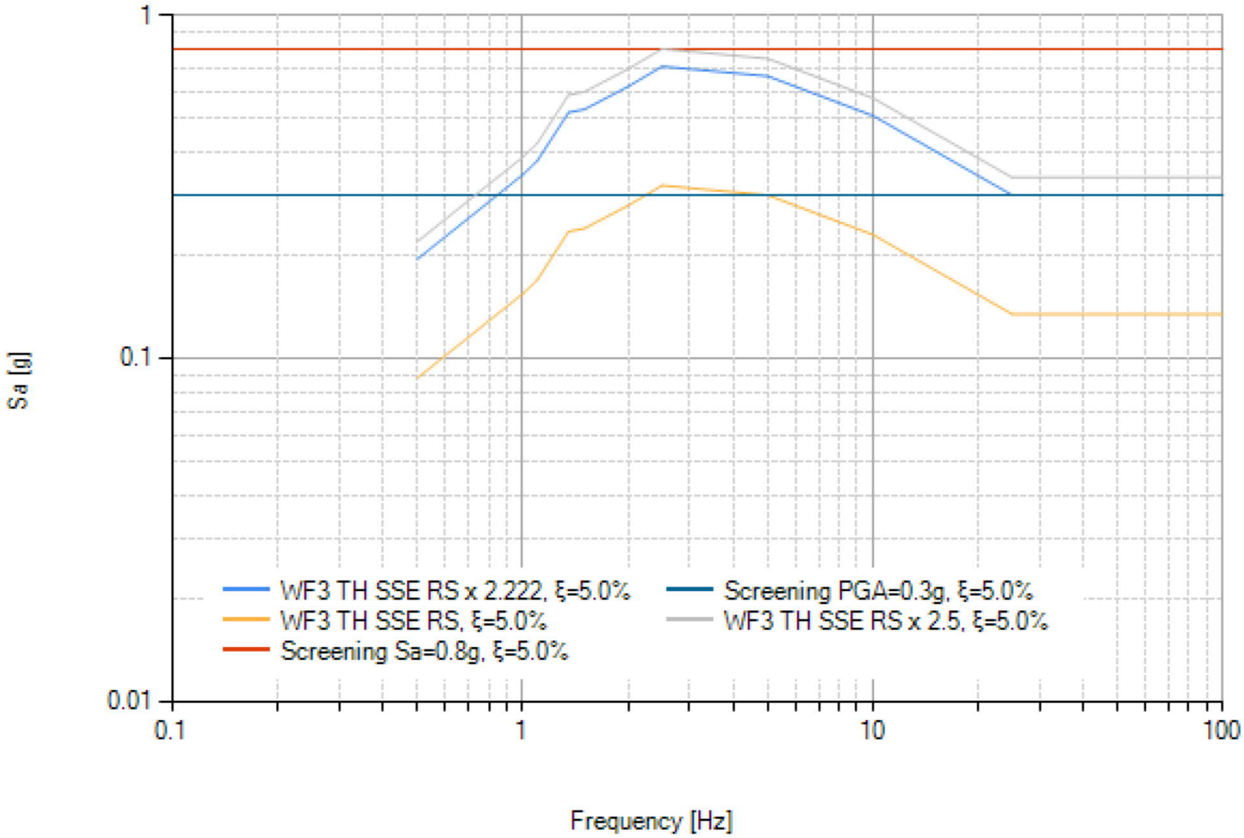


Figure 2-7 - WF3 Scaled SSE TH RS to 0.8g Screening Capacity

2.1.3.3.3 Relay Chatter

Relays will be challenging to uniformly screen out, however there are a number of factors that imply reasonably high fragilities would be found:

- All provided floor spectra in the WF3 FSAR (see Figure 2-8 for a typical example) show relatively narrow peaks which will be reduced significantly with peak clipping.
- All provided floor spectra in the WF3 FSAR, which consider variations in soil stiffness, show major peaks with the smallest period at around 0.6s which is equivalent to a frequency of 1.67 Hz. This low frequency input is not considered a concern for relays which were tested from 4-16 Hz in EPRI NP-7147-SL [42] (Seismic Ruggedness of Relays) and > 16 Hz in EPRI 3002002997 [43] (High Frequency Program - High Frequency testing Summary.)
- Minimal seismic interactions were identified during walkdowns which reduces threat of high seismic-induced accelerations resulting from localized impact between components.

Based on these considerations, it is reasonable to expect that the majority of relays and vibration-sensitive devices will have relatively high seismic margin.

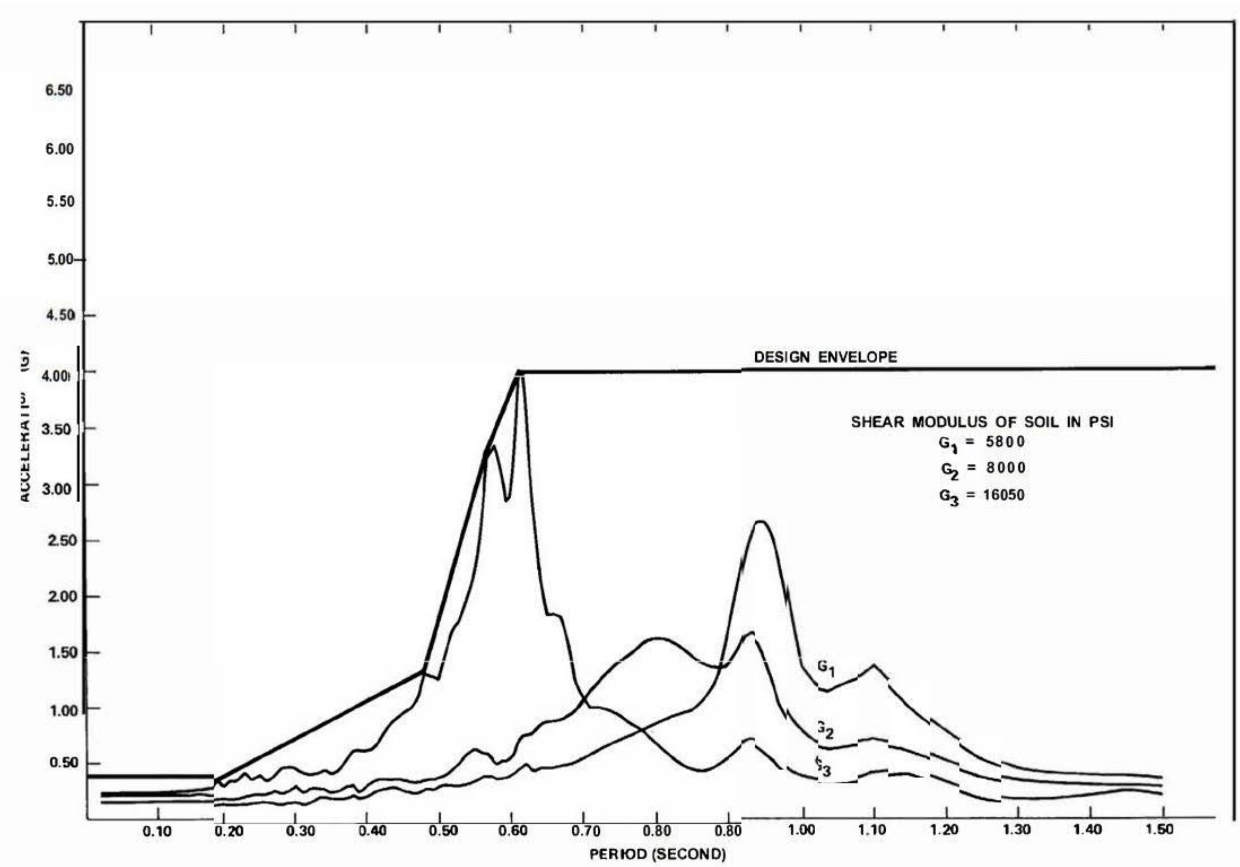
2.1.3.3.4 Anchorage and Support Evaluation

For any item that is not line-mounted, an anchorage evaluation is generally required in addition to the functional evaluation. Small and light components can often be screened out based on rugged anchorage but many components will require an anchorage evaluation to ensure the anchorage does not represent a controlling failure mode.

A typical approach for developing fragilities is the hybrid approach per 30020012994 [44], which is also referred to as the Conservative Deterministic Failure Margin method in NP-6041. In essence, the method involves identifying the capacity that would lead to an approximate 1% chance of failure and defining this as the HCLPF (High Confidence Low Probability of Failure value) and then assigning generic variability parameters to yield a fragility curve.

Figure 2-8

WF3 1% Damped E-W Floor RS for Reactor Aux. Bldg. Elev. +21 ft



In many cases, the HCLPF can be developed from the design capacity of the component being evaluated, however the two items are not the same. When performing design for components, the only requirement is for the capacity to exceed the demand. While excessive conservatism is frequently avoided, a reasonable level of conservatism is often provided to support practical design and construction. Furthermore, there are a number of load cases that must be supported in design and if the seismic load case is not governing, there is inherent conservatism in the seismic design.

In developing a HCLPF, the goal is to identify the earthquake level that matches the capacity at which failure is expected to occur 1% of the time. With this in mind, conservatism in the form of overdesign and unrealistic load cases should be eliminated in order identify the true capacity. Following typical design practice, it is assumed that there will be a reasonable safety factor, and resulting margin, incorporated into most of the seismic design of anchorage and supports.

There are also a number of additional considerations that can account for additional margin in anchorage and supports. Several of these are discussed in individual sections below.

Equipment Damping

Damping was discussed in Section 2.1.3.2 as it affects the development of ISRS through seismic response analysis. In addition to using appropriate damping for structures in the development of ISRS, it is also important to select the appropriate damping ratio for the equipment and supports in order to extract values from the ISRS. Table 3.7-1 from the WF3 FSAR, presented in Section 2.1.3.2, shows that damping ratios used for supports and assemblies are consistently lower than those recommended in RG 1.61 for SSE. Lower damping values will tend to result in higher equipment demands from ISRS for any non-rigid component, which would yield additional seismic margins in a fragility evaluation as the equipment demands would be reduced.

Ductility

Nearly all structures and components exhibit at least some ductility before failure. This yields additional seismic margin which should be accounted for. Typical factors are: 1.15 for Limit State C which accounts for limited permanent distortion and minimal damage. For equipment, this can be applied if it is clear that a ductile failure mode governs.

Concrete Strength

EPRI 3002012994 recommends a median concrete compressive strength equal to 1.5 times the minimum specified strength at 28 days and logarithmic standard deviation of 0.17 for members less than 3 ft thick when plant-specific test data are not available. This corresponds to a HCLPF-level concrete compressive strength equal to 1.13 times the minimum specified strength at 28 days (= 1.5 * exp(-1.65*0.17)).

Expansion Anchors

EPRI 3002012994 recommends a safety factor of 2.4 or 2.0 for expansion anchors. This is lower than the typical factors of safety provided by manufacturers; for example, a current Hilti Expansion Anchor has allowable loads calculated with a factor of safety of 4. A change from a safety factor of 4 to 2.4 yields a 1.67 factor increase in capacity.

2.1.3.4 Summary

The review detailed above was performed to support the assertion that 0.25g is reasonable for use as a plant level seismic HCLPF for WF3. Seismic walkdowns confirm that WF3 includes good seismic design and does not have issues with significant seismic interactions. A variety of conservatisms, in terms of both seismic demand

and seismic capacity, were identified and catalogued and indicate seismic margin beyond design-basis. While it is not possible to quantify a specific plant-level HCLPF, sufficient seismic margin has been identified to judge that a 0.25g plant-level HCLPF is reasonable.

2.1.4 Seismic Penalty Values Not Bounding

The WF3 50.69 LAR and WF3 TSTF-505 LAR inappropriately characterized the RICT seismic penalty estimates themselves as "bounding". These values are not bounding, as described below. Similar clarifications that seismic penalty SCDF and SLERF values are not "bounding" have been made in other docketed responses (e.g., refer to References 34 and 35).

Throughout the risk managed technical specifications guideline NEI 06-09 Rev 0-A and the NRC SE for that document (Reference 13), reference is made to either a "bounding" or "conservative" analysis, or sometimes to a "reasonable bounding analysis", as being acceptable to account for risk for external hazards when a PRA model is not available. The references to estimation of a seismic CDF or a seismic LERF contribution for the RICT program as "bounding" is typically inappropriate, and in the case of the WF3 TSTF-505 LAR it is inappropriate. One approach to a "bounding" estimate for SCDF would be the annual exceedance frequency of the SSE (i.e., assume a conditional core damage probability of 1.0 for SSE loads) and a "bounding" approach for estimation of SLERF would be to assume a seismic conditional LERF probability (SCLERP) of 1.0 (i.e., SCDF and SLERF are equal to the exceedance frequency of the SSS). Neither of these "bounding" approaches are reasonable or useful for risk managed technical specifications.

The estimation of seismic risk results for the WF3 RICT program are more appropriately characterized as a "conservative" analysis that uses the following:

+ Estimated SCDF
+ Estimated average SCLERP to determine an estimated SLERF, and
+ Conservative implementation of the SCDF and SLERF used in RICT assessments.

2.1.4.1 Estimated SCDF

The WF3 RICT evaluation uses an estimate of SCDF and this estimate is not bounding but is a nominal estimate with some conservative aspects in its calculation. As discussed previously, the SCDF value is determined from a mathematical convolution of the WF3 plant seismic hazard curve and an estimate of the WF3 plant level seismic fragility curve. The WF3 SCDF seismic penalty estimation approach uses convolutions of different spectral Hz curves and then employs a weighted-average approach (i.e., the "IPEEE weighted-average" approach used by the NRC in Reference 7) to estimate an SCDF from the different Hz hazard curves. The PGA-based SCDF was calculated as 1.18E-6/yr (Reference 46) but the IPEEE weighted-average value cited in the LAR (Reference 2) is an estimate almost 3x higher, 3.2E-6/yr. The WF3 seismic penalty SCDF calculation conservatively uses the PGA-based plant level fragility for the 1 Hz, 5 Hz and 10 Hz SCDF convolution calculations as well. If the PGA-based plant level fragility were adjusted per the WF3 2014 GMRS shape for the 1 Hz, 5 Hz and 10 Hz SCDF convolution calculations, the revised IPEEE-weighted average approach SCDF would be 2-3x lower than the value cited in the LAR.

2.1.4.2 Estimated SCLERP

The WF3 SLERF seismic penalty is calculated by multiplying the SCDF convolution estimate by an average seismic conditional LERF probability (SCLERP) of 0.1. The reasonableness of the 0.1 SCLERP assumed in the WF3 seismic penalty calculation is discussed below considering the following points:

+ Breakdown of SCDF by accident sequence type
+ Containment Isolation

Breakdown of SCDF by Accident Sequence Type

A given accident sequence type may not result in a core damage event until well after the PRA "Early" release time frame. Conversely, some accident sequence types would, by PRA convention, be modeled directly as a LERF release, such as seismic-induced containment failure or bypass. The contribution of various accident sequence types (or accident classes) to core damage frequency at a given plant is not necessarily the same between FPIE PRA and other hazard (e.g., seismic) PRAs.

For the purposes of presenting the reasonableness of the assumed 0.1 average SCLERP, assume that the WF3 SCDF is conservatively comprised of only three severe accident types:

+ Seismic Direct to SCDF and SLERF: Seismic induced failure of the containment function leading directly to SCDF and SLERF
+ Seismic-SBO Early: Seismic-induced station blackout with loss of all coolant makeup and heat removal at t=0 and no recovery
+ Seismic-ATWS-SBO Early: Seismic-induced coincident ATWS and station blackout with loss of all coolant makeup and heat removal at t=0 and no recovery

These three broad categories of accident sequence types are selected because they represent the scenario types with the highest CLERPs. Specific variations of these accident sequence types fall under these broader categories. For example, a Seismic-SBO Early sequence with a coincident seismic-induced LOCA would have the same CLERP as a non-LOCA Seismic-SBO Early sequence, based on review of the WF3 Level 2 PRA and assuming seismically biased constraints (e.g., no recoveries). Accident sequence types that would not produce a LERF release (e.g., Seismic-SBO-Late with steam driven systems initially operating) are conservatively removed from this assumed SCDF breakdown; this increases the weighted-average SCLERP estimate.

The contribution to SCDF of the above three accident types is calculated by first estimating the contribution of "Seismic Direct to SCDF and SLERF" and then conservatively assigning 50% of the SCDF remainder each to the other two severe accident sequence types. This is conservative given that "Seismic-ATWS-SBO-Early" scenarios have a higher CLERP than non-ATWS scenarios but they are typically much lower risk contributors to SCDF given the high seismic capacity of the reactor scram function (e.g., refer to Table 2.3 of NRC NUREG-1742 which shows that none of the IPEEE SPRAs identified failure to scram scenarios as dominant risk contributors, Reference 8).

Given that WF3 does not have a detailed plant-specific seismic PRA to assist in estimating the spectrum of SPRA accident sequence types and thus the average SCLERP, the range of containment and containment bypass fragility information from other commercial nuclear power plant SPRAs has been reviewed to assist in this breakdown. This review is summarized in Appendix A of this report. A median seismic capacity of 1.5g PGA for the containment structure, as well as for other containment bypass failure modes, is reasonably judged to be on the low end of the capacity range based on review shown in Appendix A for such fragilities.

Performing a convolution of the WF3 2014 PGA hazard curve with a 1.5g PGA median seismic capacity for containment, as well as an additional 1.5g PGA median capacity to reflect other containment bypass failure modes, both with an assumed composite beta factor (βc) of 0.4 (use of $\beta c=0.4$ in such convolution calculations in absence of plant-specific detailed fragility calculations is a commonly-accepted approximation, e.g., refer to Reference 7), results in an annual frequency of 1.8E-7/yr for "Seismic Direct to SCDF and SLERF" accident scenarios, which is approximately 6% of the LAR SCDF penalty value. As discussed previously, the remainder (94% of SCDF) is then split evenly between the other two accident scenario types.

The CLERP associated with "Seismic-SBO Early" type accidents is estimated here based on a seismically-biased sensitivity quantification of the WF3 full-power internal events (FPIE) PRA (model of record "WF3Rev6_R0") for LOOP accidents and assuming worst case conditions, i.e., SBO, no AC (offsite or onsite) recovery, no coolant makeup or decay heat removal at t=0 and no recovery of these functions. This CLERP estimate is based on quantifying the WF3 FPIE single-top WF3Rev6_R0.caf PRA model for the LERF risk metric with the following seismically-biased constraints:

+ All initiating events set to logical value of FALSE in the sensitivity quantification flag file, except the LOOP initiating event (%T5, Loss of Offsite Power) which is set to a logical value of TRUE
+ Safety DC battery basic events (DC--BAT-NO-000A, DC--BAT-NO-000B, DC--BAT-NO-00AB, DC--BAT-NO-0TGB) directly failed at t=0 by setting these failure basic events to a logical value of TRUE in the sensitivity quantification flag file
+ Emergency diesel generators DGN A, DGN B and the TEDG diesel generator directly failed at t=0 by setting the associated failure basic events to a logical value of TRUE in the sensitivity quantification flag file
+ FLEX recovery credits disabled by setting FLEX failure basic events (e.g., FLX-DGN-FS-0001, FLX-MDP-FS-0001) to a logic value of TRUE in the sensitivity quantification flag file
+ Offsite power recovery credit was disabled by adjustments in the sensitivity quantification recovery file

The resulting CLERP of the seismically-biased PRA quantification for "Seismic-SBO Early" is 4E-3. The cutsets comprising the CLERP are primarily accident progression related phenomena probabilities and a much smaller contribution from containment isolation random failure probabilities.

The same type seismically-biased PRA quantification is also made for the "Seismic-ATWS-SBO Early". This quantification run is made in the same exact manner as described above but with the additional basic event KRTMECH (FAILURE OF REACTOR TRIP (MECHANICAL)) set to a logical value of TRUE in the sensitivity quantification flag file to force the assumed ATWS condition on top of the SBO condition. The resulting CLERP of the seismically-biased PRA quantification for "Seismic-ATWS-SBO Early" is 8E-3.

The above seismically-biased CLERP values and these cutsets would be insignificantly affected by the seismic initiator as credit for mitigation functions (assumed failure of the containment function is addressed separately in the "Seismic Direct to SCDF and SLERF" accident sequence type) have already been assumed to directly fail in the calculation of these two CLERPs. The few human error probabilities remaining in these seismically-biased CLERP cutsets are non-significant contributors (e.g., even if conservatively set to an assumed 1.0 human error probability they would represent < 1% of the 4E-3 CLERP).

Using the above information, an average SCLERP of 0.06 can be calculated as follows and is less than the 0.1 SCLERP used in the WF3 TSTF-505 LAR:

SCDF Accident Sequence Type	Fraction of SCDF	SCLERP
Seismic Direct to SCDF and SLERF	0.06	1.0
Seismic-SBO Early	0.47	4E-3
Seismic-ATWS-SBO Early	0.47	8E-3
Sequence-Weighted Average SCLERP:		**0.06**

Even if the LAR SCDF penalty were revised to use scaling of the PGA-based plant level HCPLF for the other Hz convolutions (which would reduce the estimated SCDF by approximately a factor of 2x), use of the conservative

SCLERP approach above would produce a weighted average SCLERP estimate of 0.10 to 0.15 (i.e., the Seismic Direct to SCDF and SLERF contribution estimate would increase, thus increase the weighted average SCLERP estimate; refer to tab “Sect 2-1-4-2 WF3 Cont conv” in accompanying spreadsheet, Reference 49). As such, the 0.1 SCLERP used in the WF3 TSTF-505 LAR seismic penalty calculation is reasonable for the WF3 plant.

Containment Isolation

In addition to the average SCLERP estimation discussed previously, the following discussions regarding random and seismic-induced failure of containment isolation are provided to support the reasonableness of the average SCLERP estimation (e.g., there are no normally-open AC-powered MOV CIVs that would lead directly to an unscrubbed release and a LERF end state):

+ Containment Isolation Random Failure: Random failure of containment isolation is already included in the average SCLERP estimation discussion above. Random failures of containment isolation are non-significant in comparison to the 0.1 SCLERP used in the WF3 LAR SLERF penalty calculation.
+ Containment Isolation Fragility: Seismic-induced failure of containment isolation is very low likelihood and encompassed by the 0.1 average SCLERP used in the LAR. The WF3 containment isolation valves (CIVs) modeled in the WF3 Level 2 PRA containment isolation failure (CIF) fault tree and representing potential LERF release pathways are primarily air-operated valves and check valves. These are seismically-robust valves that fail-safe closed. Successful containment isolation in preventing a LERF release for seismic-induced accidents is not dependent upon pneumatic supply, electric power, or containment isolation signals.

  Check valves and AOVs are seismically rugged and as a general rule screen out of detailed SPRAs and seismic risk management applications due to high seismic capacities (e.g., refer to Section 2.3 of Reference 5).

  AOV CIVs have high seismic capacities such that seismic loading will have a negligible likelihood of failing CIVs in the open position. AOV CIVs fail-safe closed via internal spring force inside the AOV operator. Once closed, these valves do not need to open again during or after the seismic event. Therefore, they do not meet the definition of an “active” valve per the air operated valve equipment class (per the EPRI SQUG Generic Implementation Procedure, GIP, and EPRI NP-7149 Seismic Adequacy of Equipment Classes). The spring will successfully cause the CIVs to shut at accelerations much greater than those associated with the functional failure capacity used to determine the fragility of active valves. As such, these CIVs are essentially inactive valves, which are inherently rugged as there is not a credible seismic failure mechanism that would prevent the valves from failing shut as desired.

  Some containment penetrations use motor operated valves (MOV) for containment isolation which would require electric power for closure and for an isolation signal. However, such CIV MOVs are not significant to LERF for one or more of the following reasons:

  - MOV CIV in closed position during at-power operation and at the time of the seismic event
  - Very small line (non-LERF pathway)
  - AOV or check valve CIV in-series with the MOV CIV
  - Penetration is a closed-loop system or otherwise scrubbed that would not represent a LERF release.

2.1.4.3 Use of SCDF and SLERF Penalties in RICT Calculations

The majority of the conservatism in the seismic risk input to the RICT process is in the conservative use of the estimated SCDF and SLERF in RICT calculations. As appropriately described in Enclosure 4 of the WF3 TSTF-505 LAR (Reference 2), the entire annual frequencies of the calculated SCDF and SLERF penalties are applied as the seismic delta risk contribution for all RICT calculations, regardless of the duration of the RICT. Since the maximum duration for a RICT is limited to the 30-day backstop, this approach to incorporation of seismic risk insights is conservative.

A change in seismic risk exceeding the SCDF and SLERF penalties may be postulated to occur in RICT calculations; however, this is a theoretical postulation and very unlikely to occur for analyzed RICT conditions. SCDF and SLERF seismic risk as a general rule is dominated by the risk contribution from seismic fragility failure modes and seismically-based error probabilities assigned to operator actions, which are not impacted by plant changes due to the RICT program. Non-seismic failure modes (e.g., pump fails to run or start, valve fails to change position) are as a general rule of SPRAs non-significant contributors such that changes due to the RICT program are very unlikely to result in an increase in SCDF or SLERF that is equal to or greater than the SCDF and SLERF penalties used for the WF3 RICT seismic penalty calculations. Therefore, applying the total SCDF and total SLERF as delta SCDF and delta SLERF in each RICT calculation is judged to invoke sufficient conservativism regarding the incorporation of seismic risk insights into the RICT calculations.

2.2 APLC-03 PART (B) RESPONSE

The following information is provided to support that seismic risk will not solely result in a high safety significance (HSS) determination based on integrated importance measures and therefore, will not challenge the use of the qualitative consideration of seismic risk in the proposed approach.

1. <u>WF3 Hazard Meets EPRI 3002017583 Tier 1 Criteria:</u> As discussed in Section 3.2.3 of the WF3 50.69 LAR (Reference 1), the WF3 2014 seismic hazard (Reference 40) meets the low hazard (Tier 1) criteria specified in EPRI 3002017583 (Reference 5). As stated in the LAR, *"at these sites, the GMRS is either very low or within the range of the SSE such that unique seismic categorization insights are not expected."*

   As additional perspective, with respect to seismic hazard occurrence frequency the Waterford site has one of the lowest hazards in the CEUS. Refer to Figure 2-9 which shows the Mean PGA hazard occurrence frequencies from NTTF 2.1 Seismic Hazard Screening Report submittals for 44 CEUS NPP sites. If a given site used the same hazard curve for each unit then a single curve is shown; whereas if different hazard curves are used for different units on the site (e.g., Hope Creek and Salem) then separate curves are plotted. As can be seen from Figure 2-9, the Waterford site (dotted blue line), along with four other sites in low seismicity areas (Grand Gulf, South Texas Project, St. Lucie and Turkey Point, respectively as the next lowest hazard curves), is shown at the low end of the spectrum of the plotted seismic hazard curves. Hazard occurrence frequency is a direct multiplier in the seismic hazard risk calculations for a plant.

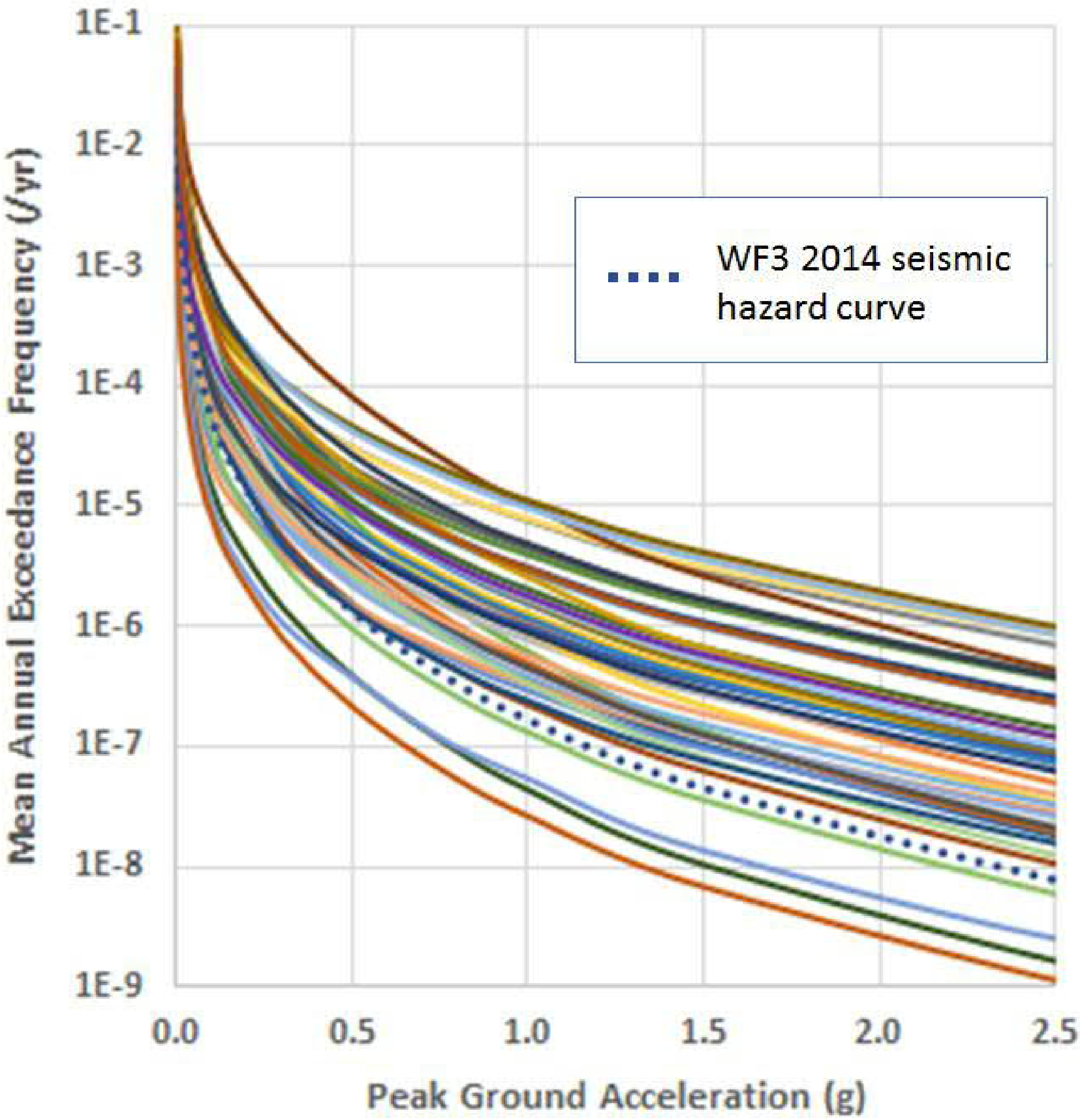


Figure 2-9 - CEUS NPP Seismic Hazards (PGA, Mean Exceedance Frequency)

2. Limited Unique Seismic Insights: The NRC is correct that Section 3.2.3 of the WF3 50.69 LAR (Reference 1) includes statements that imply that low estimated seismic risk is key to use of the EPRI 3002017583 Tier 1 seismic alternative process. However, the EPRI seismic alternative report 3002017583 (Section 2.2.2.1) does not explicitly state that the relative contribution of seismic risk is (or needs to be) low compared with the overall plant risk to justify application of the Tier 1 seismic alternative process. In fact, based on the trial studies, EPRI 3002017583 concludes that for both low hazard sites and higher hazard sites that the potential of an SSC being identified as HSS uniquely due to seismic risk calculations is low likelihood:

   "The test cases described in Section 3 showed that even for plants with high seismic ground motions compared to their design basis, there would be very few if any SSCs designated HSS for seismic unique reasons. At the low seismic hazard sites in Tier 1, the likelihood of identifying a unique seismic condition that would cause an SSC to be designated HSS is very low." (Section 2.2.2, Reference 5)
3. Conservatisms in SCDF and SLERF Penalty Estimates: As discussed previously, the SCDF and SLERF penalty estimates provided in the WF3 RICT LAR are approximations containing aspects of conservatism. In addition, the SCDF and SLERF seismic penalty approximations do not explicitly account for the following diverse accident mitigation features in place at the WF3 plant: TEDG emergency diesel generator and FLEX.
   - TEDG: The TEDG emergency diesel generator has the same power capacity as each of the two original design DGN A and DGN B emergency diesel generators, but the TEDG is diverse in design and location. This diversity in design and location means that it would not be postulated to experience a seismic-induced correlated failure (i.e., increased likelihood of all DGs failing due to similar seismic capacity and seismic response) with the other original design diesel generators. The TEDG is a skid-mounted unit that has been used during refueling outages but it has been made a permanent modification (now credited in the WF3 FPIE and FPRA PRA models). TEDG is permanently located onsite and can be aligned to either safety division.
   - FLEX: FLEX mitigating strategies were developed in response to the Fukushima accident. The FLEX strategies are designed to reduce the risk contribution for beyond design basis scenarios. FLEX equipment at WF3 is located entirely with the Reactor Aux Building (RAB), which is a seismically qualified structure. The FLEX strategy relies primarily on permanently installed equipment with portable fluid and power connections being implemented in response to the accident. These connections are not rigid and consist of portable cable reels and flexible hoses that are stored in close proximity to the component they support.
4. Significant Fraction of SCDF and SLERF Estimate Not Directly Applicable: Although not explicitly addressed and accounted for in the NEI 00-04 (Reference 4) construct for SSC categorization, a significant fraction of calculated SCDF and SLERF would not be directly applicable or useful for SSC categorization purposes. This fraction is comprised of seismic induced severe damage states (e.g., seismic-induced building failures, seismic-induced RPV support failure, seismic-induced containment failure, seismic-induced containment bypass) that are modeled in seismic risk assessments as leading directly to SCDF (and also directly to SLERF for some portion of those severe damage states). SCDF and SLERF seismic risk as a general rule is dominated by the risk contribution from seismic fragility failure modes and seismically-based error probabilities assigned to operator actions, which are not impacted by plant changes due to the RICT program. Non-seismic failure modes (e.g., pump fails to run or start, pump in test or maintenance, valve fails to change position) that would be adjusted for RICT calculations, are as a general rule non-significant contributors to SCDF or SLERF.

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4.0 Tables

5.0 Figures

Appendices

Table of Contents

Appendix A *Review of Available Industry Containment Fragility Information*

This appendix provides available industry fragility information for containments and containment bypass mechanisms for use in estimating the contribution of such seismic-induced failures to the WF3 estimated SCDF seismic penalty.

<u>Overview</u>

Given that the WF3 seismic penalty SCDF estimate is based on a convolution of a seismic hazard curve with a limiting plant HCLPF (based on the IPEEE SMA), and the fact that WF3 does not have a detailed plant-specific seismic PRA to assist in estimating the spectrum of SPRA accident sequence types, the estimation of the fraction of SCDF that results in SLERF must consider representative fragility information from available industry sources. The need for the use of representative seismic fragility information is acknowledged in numerous industry and NRC guidelines; some of these are summarized below.

Appendix H of the 2013 EPRI SPRA Implementation Guide (Reference 15) states the following regarding the use of representative seismic fragilities:

Table H-1 provides a summary of fragilities based on a survey of available industry information. The fragilities provided in Table H-2 are selected to be reasonable representative fragilities based on the assessment of industry information summarized in Table H-1.... This information is intended for use in... SPRA scoping evaluations to support risk-informed applications (such as License Amendment Request submittals to the NRC) in the absence of plant-specific seismic fragilities. ...

In selecting a reasonable representative value for the purposes of supporting an SPRA scoping evaluation (such as to support a risk-informed License Amendment Request submittal), the fragilities in Table H-2 may be used. However, the ranges of fragilities in Table H-1 should be reviewed to ensure that the selection of a reasonable representative value appropriately supports the intent of the SPRA scoping evaluations.

These representative fragilities are not intended to be conservatively low values. Plant-specific fragilities for a given SSC may result in higher or lower Am values than those provided in Table H-2....

If the analyst has a purpose for the use of conservatively low representative fragilities to highlight an issue or for other purpose (for example, regulatory requirements or guidelines require "conservative" estimates for SSCs not supported by detailed plant-specific calculations), consult the information in Table H-2 to identify ranges of values to assist in the selection of low values.

Section 2.4.2 of Volume 2 of the NRC RASP (Risk Assessment Standardization Project) handbook (Reference 47) provides the following guidance to NRC risk analysts when developing seismic risk information in the absence of available plant-specific fragility information:

The fragilities of the major SSCs must be obtained to calculate seismic failure probabilities. Preferably, the analyst should use the plant-specific fragility value if one exists for the plant. In the absence of plant-specific SSC fragilities, fragility values from power plants of similar vintage may be used as surrogates by NRC risk analysts when obtaining risk insights for operational events via the SDP, the ASP Program, Notice of Enforcement Discretion (NOED) evaluations, and event assessments under the Management Directive 8.3, "NRC Incident Investigation Program." For plant-specific risk-informed licensing applications, the fragility values should be developed by meeting the appropriate Standard and guidance.

A more extensive collection of SSC seismic fragilities is available in an NRC document (Not Publicly Available, ADAMS Accession No. ML071220070), which contains proprietary information. Many of the *values in the collection are obtained from the Individual Plant Examination of External Events (IPEEE) vintage and older compilations. In the case that plant-specific fragilities are not available, the analyst should review this collection along with more recent results to select appropriate surrogate values for the situation being analyzed. In addition, as seen from the collection, the recorded fragility values may have a wide range for a given component....*

A number of seismic PRAs are being performed in connection with the implementation of the NTTF Recommendation 2.1. These PRAs will provide a more current estimates of fragilities using the recent guidance....

These values should not be taken as NRC staff-endorsed values and the values for a specific situation should be determined using the collection of data and other relevant information as described above.

The conclusion from the above is that representative fragility information is needed in certain situations (typically when a plant SPRA is not available with plant-specific fragilities) and in such cases the analyst should consider the following: 1) range of fragility values; 2) recent fragility information; 3) available plant information to inform the selection of representative fragility values; and 4) to select representative fragility values appropriate for the risk information development at hand. Accordingly, this appendix provides a range of fragility information for the following to support the estimation of the fraction of SCDF that results in SLERF:

+ Containment structure
+ Containment bypass

The first item is the containment structure itself (such a fragility is often modeled in an SPRA directly as SCDF as well as directly as SLERF). Containment bypass considers other SSC failures (e.g., steam generator anchorage failure or RPV support failure) that are often modeled in SPRAs to directly defeat the containment function and thus directly to SCDF and SLERF. Information is obtained from the following sources:

+ UCID-20571 (Reference 48)
+ EPRI 2013 SPRA Implementation Guide (Reference 15)
+ NTTF 2.1 Seismic SPRA submittals (References 17 thru 27, 29, 31)

It is acknowledged that some of the entries from the different sources likely overlap from review of some of the same past SPRA studies. This fact does not impact the usefulness of the range of fragility values.

<u>UCID-20571 Fragility Information</u>

The UCID-20571 report (Reference 48) is a 1985 Lawrence Livermore National Laboratory compilation of SPRA fragility information from industry SPRAs that had been performed in the first half of the 1980s. The reference source cites the plant and fragility failure mode and associated fragility statistics. Some of the plants contained in the UCID-20571 report are un-named (e.g., Plant aaa, Plant bbb) for various reasons. Table A-1 provides the fragility information (Am and failure mode) from this reference source that are related to the topics of interest to this report (i.e., containment structure and containment bypass). Some of the entries in the UCID-20571 report are identified in the report to be viewed with caution for various reasons and should be discounted; such entries are not included in the Table A-1 summary here.

<u>EPRI SPRA Implementation Guide Fragility Information</u>

As discussed previously, Appendix H of the EPRI 2013 SPRA Implementation Guide (Reference 15) provides ranges of SSC fragility information based on review of various industry SPRAs. Table A-2 provides the fragility information (Am and failure mode) from this reference source that are related to the topics of interest to this report (i.e., containment structure and containment bypass). Not all the safety structure fragility information from

Appendix H was re-produced here in Table A-2, e.g., the reactor building information is not reproduced here as the WF3 containment is different in design and concept than the BWR reactor buildings represented in Appendix H of Reference 15.

NTTF 2.1 SPRA Submittal Fragility Information

Table A-3 provides the fragility information (Am and failure mode) from available NTTF 2.1 SPRA submittals (References 17 thru 27, 29, 31) that are related to the topics of interest to this report (i.e., containment structure and containment bypass). This information was compiled by accessing the submittals from the www.nrc.gov website and reviewing the information contained in the submittals. In many cases specific fragility values were not cited but could be back-calculated (e.g., a fragility basic event probability for a specific hazard interval provided in a cutset summary table) or estimated as greater than a nominal g, PGA value based on review of other information provided in the submittal.

Table A-1
Selected Fragility Information (UCID-20571)

Record #	Plant	Am (g, PGA)	Failure Mode
Containment Structural Related Fragilities			
105	Plant aaa (not identified)	2.46	Torus (support failure)
1009	Plant aaa (not identified)	1.13	Shield wall (wall shear)
106	Plant ccc (not identified)	3.08	Primary containment (wall shear)
107	Plant ccc (not identified)	5.66	Primary containment (wall flex)
108	Plant ccc (not identified)	1.35	Primary containment (wall flex)
109	Plant ddd (not identified)	3.44	Reactor building (wall shear)
111	Plant eee (not identified)	3.30	Secondary containment (wall base shear)
112	Plant eee (not identified)	9.20	Primary containment (wall base shear)
113	Plant fff (not identified)	2.00	Reactor building (bldg impact)
114	Indian Pt. 2	1.35	Containment (wall shear)
115	Indian Pt. 3	2.09	Containment (wall shear)
116	Limerick	1.97	Primary containment (wall flex)
117	Midland	2.83	Containment (wall shear)
118	Oconee U3	2.46	Containment (wall shear)
119	Millstone U3	4.90	Containment (wall shear)
120	Seabrook U1	8.20	Containment (wall shear)
121	Seabrook U1	7.60	Containment (wall flex)
123	Zion U1	4.78	Containment (sump failure)
124	Plant bbb (not identified)	1.84	Shield wall (base uplift)
125	Plant bbb (not identified)	1.60	Reactor pedestal (flexure)
126	Plant bbb (not identified)	1.72	Shield wall (wall base shear)
127	Plant ddd (not identified)	3.01	Containment Internal Structure (crane wall flex)
128	Plant eee (not identified)	5.80	Reactor pedestal (wall base shear)
129	Plant eee (not identified)	2.10	Shield wall (base anchorage)
130	Midland	2.34	Shield wall (flexure)
131	Oconee U3	5.54	Shield wall (secondary) (wall shear)
132	Oconee U3	2.83	Shield wall (primary) (wall shear)
133	Millstone U3	2.20	Containment Internal Structure (crane wall)
134	Zion U1	2.21	Pressurizer Enclosure Roof (roof collapse)
Containment Bypass Related Fragilities			
236	Plant ddd (not identified)	2.50	Steam generator (supports)
237	Indian Pt. 2	2.26	Steam generator (supports)
238	Indian Pt. 3	2.26	Steam generator (supports)
239	Midland	2.46	Steam generator (supports)
240	Seabrook U1	1.71	Steam generator (supports)
242	Zion U1	1.76	Steam generator (supports)
938	Plant aaa (not identified)	2.46	RPV support (anchor bolts)
941	Plant ccc (not identified)	1.60	RPV support (skirt)
942	Plant ddd (not identified)	2.00	RPV support

Table A-1
Selected Fragility Information (UCID-20571)

Record #	Plant	Am (g, PGA)	Failure Mode
944	Plant fff (not identified)	3.00	RPV support (skirt)
947	Limerick	1.84	RPV support (lateral support)
948	Midland	2.46	RPV support (lateral support)
949	Oconee U3	1.45	RPV support (anchor bolts)
951	Seabrook U1	2.00	RPV support

Table A-2
Selected Fragility Information (EPRI SPRA Implementation Guide)

Am (g, PGA)	*Failure Mode*
	Containment Structural Related Fragilities
1.6	BWR primary containment (shield wall shear)
1.1	BWR primary containment (shield wall star truss)
2.5	Containment (pre-stressed shear)
2.9	Containment (pre-stressed shear)
2.9	Primary containment (wall shear)
>3g	Multiple other containment fragilities in the 3-8g range
2.0	Containment - Table H-2 recommended representative value
	Containment Bypass Related Fragilities
3.02	Steam generator
1.70	Steam generator (lower supports)
2.0	Steam generator (support)
2.3	Steam generator
2.5	Steam generator (supports)
3.2	Steam generator (supports)
6.1	Steam generator (supports)
2.5	SG/Pressurizer - Table H-2 recommended representative value
1.25	RPV support (upper lateral support)
4.94	RPV support (upper support)
2.50	RPV support
2.24	RPV support
1.10	RPV support (skirt support bolts)
1.40	RPV support (skirt support bolts)
1.60	RPV support (upper lateral support)
2.30	RPV support
2.50	RPV support
4.10	RPV support (upper support)
2.0	RPV support - Table H-2 recommended representative value

Table A-3
Selected Fragility Information (NTTF 2.1 SPRA Submittals)

Plant(1)	Plant Type	SSE (g, PGA)	Containment Structural	Containment Bypass	Comment
Beaver Valley (Ref. 17)	PWR	0.12 (Ref. 7)	Am=3.37g	Am=2.71g (Steam Generator anchorage)	
Browns Ferry (Ref. 18)	Mk I BWR	0.2 (Ref. 7)	Am>2g (Note 3)	Am>2g (Break Outside Containment, Piping) (Note 3)	
Callaway (Ref. 19)	PWR	0.2 (Ref. 7)	Am=1.67g (Soil Failure)	Am=1.78g (Containment Penetration Lines)	SG anchorage failure also modeled as causing containment failure but has higher capacity (Am=1.86g) than does the fragility of containment penetration lines.
Columbia (Ref. 29)	Mk II BWR	0.25	Am=4.3g (RB/TB/RW combined structural fragility)	Am>2g (Note 3)	Note that a containment bypass scenario through RWCU due to loss of power to the isolation MOVs is also captured by the combined RB/TB/RW structural fragility.
DC Cook (Ref. 20)	PWR	0.2 (Ref. 7)	Am=1.56g	Am>2g (Note 3)	
Diablo Canyon (Ref. 31)	PWR	0.4 (Ref. 22)	Am=5.22g	Am=9.77g (Steam Generator anchorage)	Other containment bypass/failure fragilities modeled but are higher capacities than the SG fragility.
Dresden (Ref. 21)	Mk I BWR	0.2 (Ref. 7)	Am=2.35g (Drywell)	Am=2.77g (Break Outside Containment, Piping)	
North Anna (Ref. 22)	PWR	0.12 (Ref. 7)	Am=1.71g	Am>2g (Note 3)	
Peach Bottom (Ref. 23)	BWR	0.12 (Ref. 7)	Am=3.00g (Drywell)	Am=2.69g (Break Outside Containment, Piping)	

Table A-3
Selected Fragility Information (NTTF 2.1 SPRA Submittals)

Plant[(1)]	*Plant Type*	*SSE (g, PGA)*	*Containment Structural*	*Containment Bypass*	*Comment*
Sequoyah (Ref. 24)	Ice Condenser PWR	0.18 (Ref. 7)	Am>2g (Note 3)	Am-1.79g (RPV Support Failure modeled as direct CD and LERF)	SG anchorage failure also modeled as causing containment failure but has higher capacity (Am=2.24g) than does RPV supports.
VC Summer (Ref. 25)	PWR	0.15 (Ref. 7)	Am>2g (Note 3)	Am>2g (Note 3)	
Vogtle 1,2 (Ref. 26)	PWR	0.2 (Ref. 7)	Am=2.9g	Am=2.75g (Steam Generator anchorage)	
Watts Bar (Ref. 27)	Ice Condenser PWR	0.18 (Ref. 7)	Am=2.99g	Am>2g (Note 3)	

Notes to Table A-3:

(1) The NTTF 2.1 SPRA submittals for the Oconee and Robinson plants are not publicly available and are not included in this table.
(2) Information not discernable from the NTTF 2.1 SPRA submittal.
(3) Fragility values not explicitly cited in submittal, determined based on interpretation of risk results provided in Section 5 (and in some cases Appendix A of the submittal) of the NTTF 2.1 SPRA submittal.
(4) From review of the NTTF 2.1 SPRA submittal the direct to core damage probabilities are also conservatively (in some cases) modeled as leading directly to LERF.

Enclosure 3

W3F1-2021-0050

List of Regulatory Commitments

List of Regulatory Commitments

The following table identifies those actions committed to by Entergy in this document. Any other statements in this submittal are provided for information purposes and are not considered to be regulatory commitments.

Commitment	Type (check one)		Scheduled Completion Date
	One-Time Action	Continuing Compliance	
The systematic review of key assumptions and sources of uncertainty in PSA-WF3-08-06 will be updated to include more thorough documentation of the process used prior to categorization of any SSCs. This update will include a review of plant specific assumptions to verify that all key assumptions and sources of uncertainty are captured for disposition for consideration in this application. The screening process documentation will be updated to list the specific screening criteria used of determining if assumptions and sources of uncertainty that are "key" for their respective hazards are also potentially "key" for the application. The specific criteria to be used are those listed in the response to APLA 2.c. Additionally, the documentation update will include the aspects of NUREG 1855 that are employed. This update will denote the FLEX system as a key source of uncertainty for this application	✓		Prior to categorization, December 2021
The next Fire PRA model revision will include an update to ignition frequencies.	✓		December 2023
If any of the systems (EFW, DC, ID, SSD, 4KV) for which the sensitivity study shows changes in the safety significance in the fire PRA model are selected for categorization prior to the update of the ignition frequencies to the industry consensus approach (currently NUREG 2169), the results will be shared with IDP members during review to ensure they are both aware of the model limitations, but also that these limitations are related to a limited subset of components.	✓		December 2023

Commitment	Type (check one)		Scheduled Completion Date
	One-Time Action	Continuing Compliance	
The results of the FLEX equipment sensitivity study will be shared with the 10 CFR 50.69 Integrated Decision-making Panel (IDP) for categorization of systems shown to have changes in safety significance (EDG, EFW, and FLEX) to ensure they are aware of the impact of FLEX modeling when making decisions. The IDP will have the list of impacted components and systems to allow for informed decision-making for all 10 CFR 50.69 categorization efforts.		✓	December 2021
In the section "Monitoring of Inputs to and Outcome of Proposed Alternative Seismic Approach" of the CCNPP SE, the configuration control program for CCNPP had been updated to include a checklist of configuration activities to recognize those systems that have been categorized in accordance with 10 CFR 50.69, to ensure that any physical change to the plant or change to plant documents is evaluated prior to implementing those changes. This checklist is the same as what is included in Section 3.5 of the Waterford 3 LAR except for "Review of impact to seismic loading and SSE seismic requirements, as well as the method of combining seismic components." This checklist item will also be included in the Entergy configuration control program.	✓		December 2021