From: Drake, Jason
Sent: Monday, July 26, 2021 4:50 PM
To: Devoe, Remy
Cc: mthigpe@entergy.com; wjohn13@entergy.com; cmahan@entergy.com; Gaston, Ronald William; aharris@entergy.com; Dixon-Herrity, Jennifer; Buckberg, Perry
Subject: Final RAIs to Entergy Operations, Waterford Steam Electric Station, Unit 3 LAR to Adopt 10 CFR 50.69
Attachments: APLC Final RAIs - Waterford 50.69 7-26-21.pdf; APLA Final RAIs - Waterford 50.69 7-26-21.pdf

Remy,

As discussed last week, final edited RAIs are being sent per your request. The minor editorial to APLA RAI-04, Section iii (a) has been incorporated. No changes were made to the APLC RAIs from draft form since no clarifications or edits were noted. The request for 45 days to respond has been granted but is considered initiated as of the July 7th clarification call as no technical content has changed from the draft versions.

Please call me if you have any comments or concerns.

Regards,

Jason Drake
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Division of Operating Reactor Licensing
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Hearing Identifier: NRR_DRMA
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REQUEST FOR ADDITIONAL INFORMATION REGARDING THE

WATERFORD STEAM ELECTRIC STATION, UNIT 3

LICENSE AMENDMENT REQUEST TO ADOPT 10 CFR 50.69

EPID L-2020-LLA-0279

DOCKET NO. 50-382

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.

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>APLC Question 03 – Seismic Risk Contribution</u>

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.

APLC Question 04 – Change of Seismic Hazards

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.

PROBABILISTIC RISK ASSESSMENT LICENSING BRANCH A

REQUEST FOR ADDITIONAL INFORMATION

REGARDING THE

WATERFORD STEAM ELECTRIC STATION, UNIT 3

LICENSE AMENDMENT REQUEST TO ADOPT 10 CFR 50.69

EPID L-2020-LLA-0279

DOCKET NO. 50-382

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

[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 Mechnical 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).

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 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.

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

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[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).

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

[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).

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-1855[7] provides acceptable guidance for the treatment of uncertainties in risk-informed decisionmaking.

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.

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

[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)

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.

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):

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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."

> 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, 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:

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

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.

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 dependency analysis prior to implementation of the 10 CFR 50.69 categorization.

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