UNITED STATES
NUCLEAR REGULATORY COMMISSION
WASHINGTON, D.C. 20555-0001

October 7, 2021

Site Vice President
Entergy Operations, Inc.
Waterford Steam Electric Station, Unit 3
17265 River Road
Killona, LA 70057-3093

SUBJECT: WATERFORD STEAM ELECTRIC STATION, UNIT 3 – REGULATORY AUDIT IN SUPPORT OF LICENSE AMENDMENT REQUEST TO REVISE TECHNICAL SPECIFICATIONS TO ADOPT RISK-INFORMED COMPLETION TIMES (EPID L-2021-LLA-0014)

Dear Sir or Madam:

By letter dated February 8, 2021 (Agencywide Documents Access and Management System (ADAMS) Accession No. ML21039A648), Entergy Operations, Inc. (Entergy, the licensee) submitted a license amendment request for the Waterford Steam Electric Station, Unit 3 to adopt Technical Specifications Task Force (TSTF) Traveler 505 (TSTF-505), "Provide Risk-informed Extended Completion Times – RITSTF [Risk Informed TSTF] Initiative 4b," to permit the use of risk-informed technical specification completion times for certain actions required when limiting conditions for operation are not met.

The U.S. Nuclear Regulatory Commission (NRC) staff has reviewed the licensee's application and determined that a regulatory audit would assist in the timely completion of the review. The NRC staff will conduct a regulatory audit to support its review in accordance with the enclosed audit plan. A regulatory audit is a planned activity that includes the examination and evaluation of primarily nondocketed information.

The NRC staff will conduct the audit virtually at NRC Headquarters in Rockville, Maryland, via a licensee-established electronic portal available to NRC staff from approximately July 19, 2021, through November 30, 2021, with formal audit meetings from October 12, 2021, through October 14, 2021. The audit plan, which was discussed with your staff on September 29, 2021, is enclosed with this letter.

If you have any questions, please contact me at (301) 415-8378 or by e-mail at Jason.Drake@nrc.gov.

Sincerely,

/RA/

Jason Drake, Project Manager
Plant Licensing Branch IV
Division of Operating Reactor Licensing
Office of Nuclear Reactor Regulation

Docket No. 50-382

Enclosure:
Audit Plan

cc: Listserv

REGULATORY AUDIT PLAN

BY THE OFFICE OF NUCLEAR REACTOR REGULATION

TO SUPPORT THE REVIEW OF LICENSE AMENDMENT REQUEST TO REVISE

TECHNICAL SPECIFICATIONS TO ADOPT RISK-INFORMED COMPLETION TIMES

ENTERGY OPERATIONS, INC.

WATERFORD STEAM ELECTRIC STATION, UNIT 3

DOCKET NO. 50-382

1.0 BACKGROUND

By letter W3F1-2021-0003 dated February 8, 2021 (Agencywide Documents Access and Management System (ADAMS) Accession No. ML21039A648), Entergy Operations, Inc. (Entergy, the licensee) submitted a license amendment request (LAR) for the Waterford Steam Electric Station, Unit 3 (Waterford 3 or WF3).

The proposed amendment would modify technical specification (TS) requirements to permit the use of risk-informed completion times (RICTs) with the implementation of Nuclear Energy Institute Topical Report (TR) (NEI) 06-09, Revision 0-A "Risk-Informed Technical Specifications Initiative 4b, Risk-Managed Technical Specifications (RMTS) Guidelines," dated November 2006 (ADAMS Accession No. 12286A322). The proposed changes are based on Technical Specifications Task Force (TSTF) Traveler TSTF-505, Revision 2, "Provide Risk Informed Extended Completion Time – RITSTF [Risk Informed TSTF] Initiative 4b, dated July 2, 2018 (ADAMS Accession No. ML18183A493).

2.0 REGULATORY AUDIT BASES

A regulatory audit is a planned licensing or regulation-related activity that includes the examination and evaluation of primarily nondocketed information. The audit is conducted with the intent to gain understanding, to verify information, and to identify information that will require docketing to support the basis of a licensing or regulatory decision. Performing a regulatory audit is expected to assist the U.S. Nuclear Regulatory Commission (NRC) staff in efficiently conducting its review of the LAR and to gain insights to the licensee's processes and procedures. Information that the NRC staff relies upon to make the safety determination must be submitted on the docket.

The basis of this audit is Entergy's LAR for Waterford 3, and NUREG-0800, "Standard Review Pan for the Review of Safety Analysis Reports for Nuclear Power Plants: LWR [Light-Water Reactor] Edition," Section 19.2, "Review of Risk Information Used to Support Permanent Plant-Specific Changes to the Licensing Basis: General Guidance," dated June 2007 (ADAMS Accession No. ML071700658).

This audit will be conducted in accordance with the Office of Nuclear Reactor Regulation (NRR) Office Instruction LIC-111, Revision 1, "Regulatory Audits," dated October 2019 (ADAMS Accession No. ML19226A274). An audit was determined to be the most efficient approach toward a timely resolution of issues associated with this LAR review, since the NRC staff will have an opportunity to minimize the potential for multiple rounds of requests for additional information and ensure no unnecessary burden will be imposed by requiring the licensee to address issues that are no longer necessary to make a safety determination.

3.0 PURPOSE AND SCOPE

The purpose of the audit is to gain a more detailed understanding of the licensee's process to implement TSTF-505 and how it conforms to NRC-endorsed guidance in TR NEI 06-09, Revision 0-A. The NRC staff will audit the probabilistic risk assessment (PRA) methods that the licensee would use to determine the risk impact from which the revised completion times would be obtained, including the licensee assessments of internal events (including internal flooding) and fire PRAs.

The areas of focus for the regulatory audit are the information contained in the licensee's submittal, the enclosed audit information needs, and all associated and relevant supporting documentation (e.g., methodology, process information, calculations, etc.) identified below.

4.0 INFORMATION AND OTHER MATERIAL NECESSARY FOR THE AUDIT

The NRC staff will request information and interviews throughout the audit period. The NRC staff will use an "audit items list" to identify the information (e.g., methodology, process information, and calculations) to be audited and the subjects of requested interviews and meetings. The NRC staff will provide the final audit items list as an enclosure to the audit summary report, which will be publicly available. The attachment to this audit plan includes the initial audit items list. Throughout the audit, the NRC staff will supplement this list with audit questions and audit-related requests so that the licensee can better prepare for audit discussions with NRC staff. Any information accessed through the licensee's portal will not be held or retained in any way by NRC staff. The NRC will use the audit items list to support the audit with the licensee, which has been scheduled for October 12, 2021, through October 14, 2021. The NRC staff requests the licensee to have the requested audit information listed in the audit items list to be readily available and accessible for the NRC staff's review via a Web-based portal.

The following documentation should be available to the audit team:

1. Applicable peer review reports and closure reports for internal events, internal flooding, and fire PRAs;

2. Uncertainty notebooks for internal events, internal flooding, and fire PRAs related to PRA model assumptions and sources of uncertainty;

3. Plant-specific documentation (e.g., uncertainty notebooks) related to the review of the PRA model assumptions and sources of uncertainty and identification of key assumptions and sources of uncertainty for the application, for the internal events, internal flooding, and fire PRA;

4. PRA notebooks for the modeling of Diverse and Flexible Mitigation Capability (FLEX) equipment and FLEX human error probabilities;

5. Documentation related to the disposition of open facts and observations;

6. Documentation supporting the example RICT calculations presented in LAR, Enclosure 1, Table E1-2, “In Scope TS/LCO [Limiting Condition for Operation] Conditions RICT Estimate”;

7. Any draft or final RICT program procedures;

8. Documentation supporting the development of the real-time risk tool and benchmarking it against the PRA; and

9. Procedure EN-DC-151, “PSA [Probabilistic Safety Analysis] Maintenance and Update.”

In addition, the following presentations and/or breakout sessions are requested:

RICT Program Presentation

1. Configuration risk management program (CRMP) demonstration (including presentation of user interface for evaluations.

2. Walkthrough sample RICT calculations.

3. Discussion on how risk management actions are determined and implemented.

4. Discuss reviews and acceptance testing of the CRMP model.

5. Discussion on how the CRMP is maintained consistent with the baseline PRA model.

6. Discuss how cumulative risk will be evaluated and tracked.

7. The RICT estimates in Table E1-2 include estimates that are less than the backstop completion time for the associated Condition. The NRC staff would like to discuss the assumptions and analysis approach that went into calculating these estimated RICTs.

5.0 AUDIT TEAM

The audit will be conducted by NRC staff from NRR, Division of Operating Reactor Licensing (DORL), Plant Licensing Branch 4 (LPL4); Division of Safety Systems (DSS), Containment and Plant Systems Branch (SCPB), Nuclear Systems Performance Branch (SNSB), and Technical Specifications Branch (STSB); Division of Engineering and External Hazards (DEX), Electrical Engineering Branch (EEEB), Instrumentation and Controls Branch (EICB), and Mechanical Engineering and Inservice Testing Branch (EMIB); Division of Risk Assessment (DRA), Probabilistic Risk Assessment Licensing Branches A , B, and C (APLA, APLB and APLC); Division of Reactor Oversight (DRO), Operator Licensing and Human Factors Branch (IOLB); Division of New and Renewed Licenses (DNRL), Vessels and Internals Branch (NVIB) and Piping and Head Penetrations Branch (NPHP) , and an NRC contractor from Pacific Northwest National Laboratory (PNNL). Key licensee personnel involved in the development of the LAR should be made available for interactions on a mutually agreeable schedule to respond to any questions from the NRC staff.

NRC Staff Team

- Jason Drake, NRR/DORL/LPL4
- Bhagwat Jain, NRR/DORL/LPL4
- Mihaela Biro, NRR/DRA/APLA
- Jonathan Evans, NRR/DRA/APLA
- Bernard Grenier, NRR/DRA/APLB
- Steve Alferink, NRR/DRA/APLC
- De Wu, NRR/DRA/APLC
- Edmund Kleeh, NRR/DEX/EEEB
- Sheila Ray, NRR/DEX/EEEB
- Ming Li, NRR/DEX/EICB
- Gursharan Singh, NRR/DEX/EICB
- Yuken Wong, NRR/DEX/EMIB
- David Nold, NRR/DSS/SCPB
- Steve Jones, NRR/DSS/SCPB
- Fred Forsaty, NRR/DSS/SNSB
- Khadijah West, NRR/DSS/STSB
- Joshua Wilson, NRR/DSS/STSB
- Dabin Ki, NRR/DRO/IOLB
- Joel Jenkins, NRR/DNRL/NVIB
- Stephen Cumblidge, NRR/DNRL/NPHP
- Mark Wilk, PNNL

6.0 LOGISTICS

The NRC staff requests the licensee to have the information discussed in Section 4.0 readily available and accessible for the NRC staff's review via an internet-based portal. The audit will occur via the internet-based portal at NRC Headquarters Office in Rockville, Maryland, from approximately July 19, 2021, through September 17, 2021, with formal audit meetings from October 12, 2021, through October 14, 2021. The NRC staff requests the licensee to have its staff generally available by telephone at mutually agreeable times during regular business hours (e.g., Monday–Thursday, 9:00 a.m. to 4:00 p.m., Eastern Time) while the internet-based portal is open from July 19, 2021, through November 30, 2021, if the NRC staff has any questions during the audit related to any information on the portal. The NRC staff requests the licensee to have its staff available by telephone and agreed virtual media during regular business hours during the scheduled formal meetings from October 12, 2021, through October 14, 2021. The NRC's licensing project manager will inform the licensee via routine communications when the NRC staff no longer needs access to the portal.

Audit Milestones and Schedule		
Activity	**Timeframe**	**Comments**
Clarification Call	September 29, 2021	Teleconference from NRC headquarters to provide clarification of audit questions.
Audit Kickoff Meeting	October 12, 2021	Brief team introduction and discussion of the scope of the audit. The licensee should introduce team members and give logistics for the week.
End of Day Summary Briefings	October 12, 2021 - October 14, 2021	Meet with licensee to provide a summary of any significant findings and requests for additional assistance.
Onsite Audit Exit Meeting	October 12, 2021	NRC staff will hold a brief exit meeting with the licensee's staff to conclude audit activities.
Audit Summary (see Section 8.0)	90 days after exit	To document the audit.

Regulatory Audit Plan Review Areas and Assignments		
	Lead	**Support**
Categorization Process	Team	Team
PRA Acceptability	APLA/APLC	PNNL
Peer Reviews	APLA/APLC	PNNL
Facts and Observations (Closure Process)	APLA/APLC	PNNL
PRA Updates/Upgrades	APLA/APLC	PNNL
External Hazards	APLA/APLC	PNNL
Integrated Decisionmaking Panel	APLA	PNNL
Documentation, Configuration Control, and Quality	APLA/APLC	PNNL
Defense-in-Depth and Safety Margins	APLA	PNNL
Risk Assessment	APLA/APLC	PNNL
Risk Management Actions	APLA	PNNL
Monitoring Program	APLA	PNNL
PRA Functionality	APLA	PNNL
PRA Technical Adequacy	APLA/APLC	PNNL
Plant-Specific Technical Specifications	DEX, DSS	

7.0 SPECIAL REQUESTS

The NRC staff would like access to the documents listed above in Section 4.0 through the online portal that allows the NRC staff and contractors to access documents via the internet.

The following conditions associated with the online portal must be maintained throughout the duration that the NRC staff and contractors have access to the online portal:

- The online portal will be password-protected, and separate passwords will be assigned to the NRC staff and contractors who are participating in the audit.
- The online portal will be sufficiently secure to prevent the NRC staff and contractors from printing, saving, downloading, or collecting any information on the online portal.
- Conditions of use of the online portal will be displayed on the login screen and will require acknowledgment by each user.
- Username and password information should be provided directly to the NRC staff and contractors. The NRC project manager will provide Entergy the names and contact information of the NRC staff and contractors who will be participating in the audit. All other communications should be coordinated through the NRC project manager.

8.0 DELIVERABLES

An audit summary, which may be public, will be prepared within 90 days of the completion of the audit. If the NRC staff identifies information during the audit that is needed to support its regulatory decision, the staff will issue requests for additional informant to the licensee after the audit.

AUDIT ITEMS LIST REGARDING

LICENSE AMENDMENT REQUEST TO REVISE TECHNICAL SPECIFICATIONS

TO ADOPT TSTF-505, REVISION 2,

"PROVIDE RISK INFORMED EXTENDED COMPLETION TIMES – RITSTF INITIATIVE 4B"

WATERFORD STEAM ELECTRIC STATION, UNIT 3

DOCKET NO. 50-382

Division of Risk Assessment (DRA)
Probabilistic Risk Assessment (PRA) Licensing Branch A (APLA) Questions

APLA Question 01 – Open Internal Events PRA Facts and Observations

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 PRA. The primary results of a peer review are the F&Os recorded by the peer review team and the subsequent resolution of these facts and observations (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 U.S. Nuclear Regulatory Commission (NRC) in a letter dated May 3, 2017.[4]

a) Section 1-A.2 of the 2009 ASME/ANS PRA Standard defines a PRA upgrade as a method new to the PRA model, and Example 24 of the Non-mandatory Appendix states that a new human reliability analysis (HRA) approach would constitute a PRA upgrade.

Table E2-1, "Waterford 3 [Waterford Steam Electric Station, Unit 3 or WF3] PRA Peer Reviews," in Enclosure 2, "Information Supporting Consistency with Regulatory Guide 1.200, Revision 2," to the license amendment request (LAR) 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

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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, dated March 2009 (Agencywide Documents Access and Management System (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," dated February 2009, New York, NY (Copyright).
[3] Anderson, V. K., Nuclear Energy Institute, letter to Stacey Rosenberg, U.S. Nuclear Regulatory Commission, "Final Revision of Appendix X to NEI 05-04/07-12/12-16, Close-Out of Facts and - Observations (F&Os)," dated 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).

the F&O closure review team as partially resolved based on the updates to the HRA spreadsheets. In the original peer review, the team stated that the licensee's analysis was "unjustified" and "inaccurate" in some places while the independent assessment (IA) team concluded that the licensee's spreadsheets "lacked thorough detail and references." Effectively, the original HRA spreadsheets appear to lack an appropriate level of justification and detail to support the review. In addition, both dispositions state that the Waterford 3 HRA was subsequently included in the use of the Electric Power Research Institute (EPRI) HRA calculator to perform the HRA. The NRC staff notes that the HRA calculator has the following HRA methods and inputs: Human Cognitive Reliability (HCR)/Operator Reliability Experiments (ORE) Method, Cause-Based Decision Tree Method (CBDTM), Performance Shaping Factors (PSFs), and stress levels in addition to Accident Sequence Evaluation Program (ASEP) HRA Procedure, and Technique for Human Error Rate Prediction (THERP). It is unclear to the NRC staff what HRA methods were used in the spreadsheets and how it evolved into the HRA calculator.[5] In light of these observations:

i. Provide a detailed comparison, describing the HRA methods used in the HRA spreadsheets and the HRA calculator.

ii. Explain how the HRA methods utilized in the HRA calculator does not constitute a PRA upgrade from the HRA spreadsheet as defined in the ASME/ANS 2009 PRA Standard.

iii. Alternatively, to Item ii above, propose a mechanism to ensure a focused-scope HRA peer review is conducted on the upgraded HRA methods and all associated F&Os closed by the Appendix X approved process prior to implementing the risk-informed completion time (RICT) program.

APLA Question 02 – 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 Diverse and Flexible Mitigation Capability (FLEX) equipment into a PRA model in support of risk-informed decisionmaking in accordance with the guidance of RG 1.200.

Regarding equipment failure probability in the May 30, 2017, memorandum, 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.

Regarding HRA, NEI 16-06, "Crediting Mitigating Strategies in Risk-Informed Decision Making," Section 7.5, "Human Reliability Assessment," recognizes that the current HRA methods do not translate directly to human actions required for implementing mitigating strategies.

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

Sections 7.5.4, “Addressing the Actions Not Currently Addressed by Existing HRA Tools,” and 7.5.5, “Addressing Complex Actions in Mitigating Strategies,” 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, memorandum, the NRC staff concluded, in part (Conclusion 11):

> Until gaps in the human reliability analysis methodologies are addressed by improved industry guidance, HEPs [human error probabilities] 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.

Item No. 1 of Table E9-1, “Internal Events/Internal Flooding PRA Assumptions & Sources of Uncertainty,” in Enclosure 9, “Key Assumptions and Sources of Uncertainty,” to the LAR, identifies the incorporation of FLEX strategies and equipment in the PRA model as a source of uncertainty and performed a sensitivity study that demonstrated this model addition had an impact on station blackout risk. The results of the study[6] demonstrate the FLEX credit decreases core damage frequency (CDF) by 7 percent. The disposition states that the inclusion of FLEX is not a key source of uncertainty since it reflects the as-built, as-operated plant. The NRC staff notes that the concern is with regard to the methods used to determine the failure probabilities for FLEX equipment and operator actions. Provide the following information for the PRA model, as appropriate:

a) LAR Enclosure 2 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 RICT calculations in accordance with ASME/ANS RA-Sa-2009, as endorsed by RG 1.200.

b) A discussion detailing the methodology used to assess operator actions related to FLEX equipment, and the licensee personnel that perform these actions. The discussion should include:

    i. A summary of how the licensee evaluated the impact of the plant-specific HEPs 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 the FLEX pre-initiators evaluation, address the following:

        (1) 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.

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

(2) Alternatively, to Item ii.(1) above, propose a mechanism to ensure incorporation of pre-initiator human failures in the PRA model prior to implementation of the RICT program.

iii. Regarding FLEX strategy initiations, address the following:

(1) If the licensee's procedures governing the initiation or entry into mitigating strategies are ambiguous, vague, or not explicit, a discussion detailing the technical bases for probability of failure to initiate mitigating strategies.

(2) Alternatively, to Item iii.(1) above, propose a mechanism to ensure entry into FLEX strategies is appropriately addressed and incorporated into the PRA model prior to the implementation of the RICT program.

c) Based on the Waterford 3 PRA documentation audited[7] 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. Provide clarification whether the Waterford 3 HRA dependency analysis process was performed utilizing the HRA calculator tools including the Dependency Decision Tree.

iii. Alternative to Items c) i and ii above, propose a mechanism to include the FLEX actions in the PRA HRA dependency analysis prior to implementation of the RICT program.

<u>APLA QUESTION 03 – PRA Model Update Process</u>

Section 2.3.4, "PRA Technical Adequacy," of Topical Report (TR) NEI 06-09, Revision 0-A, "Risk-Informed Technical Specifications Initiative 4b, Risk-Managed Technical Specifications (RMTS) Guidelines," dated November 2006 (ADAMS Accession No. 12286A322), states, in part, that "criteria shall exist in PRA configuration risk management to require PRA model updates concurrent with implementation of facility changes that significantly impact RICT calculations."

LAR Enclosure 7, "PRA Model Update Process," states that if a plant change or a discovered condition is identified and has significant impact on the RICT calculations then an unscheduled update of the PRA models will be implemented. More specifically, the LAR states that if the plant changes meet specific criteria defined in the plant PRA and update procedures, including criteria associated with consideration of the cumulative risk impact, then the change will be incorporated into applicable PRA models without waiting for the next periodic PRA update. The LAR does not explain under what conditions an unscheduled update of the PRA model will be performed or the criteria defined in the plant procedures that will be used to initiate the update. Therefore, describe the conditions under which an unscheduled PRA update (i.e., more than

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

once every two refueling cycles) would be performed, and the criteria that would be used to require a PRA update. In the response, define what is meant by "significant impact to the RICT Program calculations."

APLA QUESTION 04 – System and Surrogate Modeling Used in the PRA Models

The NRC staff's safety evaluation (SE) to TR NEI 06-09 (ADAMS Accession Nos. ML071200238 and ML122860402) specifies that the LAR should provide a comparison of the technical specification (TS) functions to the PRA modeled functions, and that justification be provided to show the scope of the PRA model is consistent with the licensing basis assumptions. Table E1-1 in Enclosure 1, "List of Revised Required Actions to Corresponding PRA Functions," to the LAR identifies each TS limiting condition for operation (LCO) proposed to be included in the RICT program and describes how the systems and components covered in the TS LCO are implicitly or explicitly modeled in the PRA. For certain TS LCO conditions, the table explains that the associated structures, systems, and components (SSCs) are not modeled in the PRAs but will be conservatively represented using a surrogate event. For some LCOs, the LAR did not provide enough description of the surrogate PRA modeling that will be used in the RICT calculations for NRC staff to understand whether the modeling will be acceptable. Therefore, address the following:

a) LAR Table E1-1, states that for TS LCO 3.6.3, "Containment Isolation Valves," not all containment isolation valves/lines are modeled in detail in the PRA, and therefore, bounding surrogates "can" be used to ensure the affected components are included in the CDF and large early release frequency (LERF) calculations. The meaning of the phrase "'bounding can be used" is not clear to the NRC staff. It is not clear to NRC staff (1) how the use of the surrogate modeling bounds the function of SSCs that are taken out of service during an RICT for this TC LCO Condition and (2) will surrogates be used in all case.

    i. Explain what component failures will be used as a surrogate to model TS LCO 3.6.3, given that not all containment isolation valves are modeled in detail in the PRA.

    ii. Describe what surrogates will be used in the model and explain the basis for the surrogate that will be used. Include an explanation of how this relates to the design basis success criterion for the containment isolation function.

b) LAR Table E1-1, states for TS LCO 3.3.2 (The Engineered Safety Features Actuation System (ESFAS) Instrumentation), Condition 7(e) (Emergency Feedwater Actuation (EFAS)), that the control valve logic is not modeled, and that other steam generator (SG) logic and other instruments will be used as a surrogate for this LCO. It is not clear how the SG logic and other instruments will be used as a surrogate for this TS LCO Condition.

   Provide details on how the surrogate will be used to model TS LCO Condition 3.3.2.7e. Include in the discussion how the surrogate is equivalent or bounding of the LCO function.

c) Regarding TS LCO 3.4.3.1, "Pressurizer," the LAR states that the pressurizer heaters are included in the PRA model with the "fail to de-energize function," which contributes to high pressurizer pressure. The LAR recognizes that the PRA model does not capture the TS function, which requires the heaters to energize when needed. To capture the risk due to entering this LCO, the LAR proposes to use a surrogate event of a stuck open atmospheric dump valve. It is unclear to the NRC staff how a stuck open atmospheric dump valve would lead to similar accident scenarios as an inoperable pressurizer heater.

Provide further justification why the proposed surrogate for LCO 3.4.3.1 is a conservative and bounding PRA treatment for the loss of the heaters.

d) Regarding LCO 3.7.6.1, "Control Room Emergency Air Filtration System," and LCO 3.7.6.3, Control Room Air Temperature – Operating," LAR Table E1-1 states: "[a]ll the components in the MCR [main control room] HVAC [heating, ventilation, and air conditioning]/filtration system are not in the PRA model, but several AHUs [air handling units], fans and dampers are included allowing for a bounding evaluation using the CRMP [Configuration Risk Management Program]."

Regarding unmodeled SSCs, the NRC staff's SE for TR NEI 06-09 states, in part:

> TR NEI 06-09, Revision 0, specifically applies the RMTS only to those SSCs which mitigate core damage or large early releases. Where the SSC is not modeled in the PRA, and its impact cannot otherwise be quantified using conservative or bounding approaches, the RMTS are not applicable, and the existing frontstop CT [completion time] would apply.

Further, Item 11 in Section 2.3, Scope," of Technical Specifications Task Force (TSTF) Traveler TSTF-505, Revision 2, states:

> The traveler will not modify Required Actions for systems that do not affect core damage frequency (CDF) or large early release frequency (LERF) or for which a RICT cannot be quantitatively determined.

Provide further justification on the proposed PRA modeling to capture the impact on CDF and LERF from entering LCOs 3.7.6.1 and 3.7.6.3, since it is only partially modeled in the PRA.

APLA QUESTION 05 – TSTF-505 – Instrumentation and Controls

TR NEI 06-09, Revision 0-A, concerning the quality of the PRA model, states that "RG 1.174, Revision 1, "An Approach for Using Probabilistic Risk Assessment in Risk-Informed Decisions on Plant-Specific Changes to the Licensing Basis," dated November 2002 (ADAMS Accession No. ML023240437) and RG 1.200, Revision 1, "An Approach for Determining the Technical Adequacy of Probabilistic Risk Assessment Results for Risk-Informed Activities," dated January 2007 (ADAMS Accession No. ML070240001) define the quality of the PRA in terms of its scope, level of detail, and technical adequacy. The quality must be compatible with the safety implications of the proposed TS change and the role the PRA plays in justifying the change."

For several TS LCO Conditions listed in LAR Table E1-1, the table indicates that instrumentation and control (I&C) modeling in PRA models is insufficient to model the Condition, and so, the inoperability of the associated SSC (e.g., channel) will be modeled using a surrogate event. For other TS LCO Conditions in the RICT program, it is not clear to NRC staff whether I&C is always modeled in sufficient detail to support implementation of TSTF-505, Revision 2 based on documentation in the LAR. The following additional information is requested:

a) Explain how I&C is modeled in the PRA. This should include (1) the scope of the I&C equipment that is explicitly included (e.g., bistables, relays, sensors, integrated circuit cards),(2) a description of the level of detail that is modeled (e.g., are all channels of an actuation circuit modeled), (3) discussion of what data and whether plant specific data is used, and (4) discussion of the associated TS functions for which an RICT can be applied.

b) Regarding LCO 3.3.2, LAR Table E1-1 states that "The Containment High Pressure and Pressurizer Pressure Low instrumentation/function will not have an RICT." Explain further the rationale for this differentiation and how will this be captured in the TS.

c) Section 2.3.4 of TR NEI 06-09, Revision 0-A, states that PRA modeling uncertainties be considered in application of the PRA base model results to the RICT program. The NRC SE for TR NEI 06-09, Revision 0, states that this consideration is consistent with Section 2.3.5 of RG 1.177, Revision 1. The guidance in TR NEI 06-09, Revision 0-A, further states that sensitivity studies should be performed on the base model prior to initial implementation of the RICT program on uncertainties, which could potentially impact the results of an RICT calculation, and that sensitivity studies should be used to develop appropriate compensatory risk management actions (RMAs).

   Regarding digital I&C, known modeling challenges exist such as the lack of industry data for digital I&C components, the difference between digital and analog system failure modes, and the complexities associated with modeling software failures including common cause software failures. Also, though reliability data from vendor tests may be available, this source of data is not a substitute for in-the-field operational data. Given these challenges, the uncertainty associated with modeling a digital I&C system could impact the RICT program. It Is not clear to NRC if there are digital system that are credited in the PRA models that will be used in the RICT program.

   i. Confirm that no digital I&C systems are credited in the PRA models that will be used in the RICT program.

   ii. If other digital I&C systems are credited in the PRA models that will be used in the RICT program, then:

      1. Identify those systems and provide the results of a sensitivity study on the SSCs in the RICT program demonstrating that the uncertainty associated with modeling digital I&C systems has an inconsequential impact on the RICT calculations.

      2. Alternatively, identify which LCOs are determined to be impacted by the digital I&C system modeling for which RMAs will be applied during an RICT. Explain and justify the criteria used to determine what level of impact to the RICT calculation require additional RMAs.

<u>APLA QUESTION 06 – Potential Loss of Function Conditions</u>

The model SE of TSTF-505 Revision 2 (ADAMS Accession No. ML17290A005), limited applicability of the TSTF to conditions that were not considered TS loss of function (LOF).

Additionally, the licensee did not propose any constraints within its administrative controls, which would allow for LOF. Therefore, TS LOF conditions are not in the scope of this application. For LCO 3.7.1.5, with one Main Steam Isolation Valve (MSIV) inoperable, LAR Table E1-1 indicates the design criteria is that both MSIVs close. Based on this information and Revision 2 to TSTF-505, it appears that LCO 3.7.1.5 could constitute an LOF, and therefore would be not covered by TSTF-505 Revision 2. Justify why these conditions can be subject to an RICT.

DRA PRA Licensing Branch C (APLC) Questions

APLC Question 01 – High Confidence of Low Probability of Failure

As clarified in the NRC SE on TR NEI 06-09, Revision 0-A, other sources of risk (i.e., seismic and other external events) must be quantitatively assessed if they contribute significantly to configuration-specific risk. The SE on TR NEI 06-09, Revision 0-A, also states that bounding analyses or other conservative quantitative evaluations are permitted where realistic PRA models are unavailable.

LAR Enclosure 4, “Information Supporting Justification of Excluding Sources of Risk Not Addressed by the PRA Models,” Table E4-1, “External Hazard Evaluation,” presents the licensee’s bounding analysis of seismic risk to Waterford 3. The LAR states that a convolution was performed with a high confidence of low probability of failure (HCLPF) value of 0.25g. Engineering Report PSA-WF3-04-01, “Seismic Risk Evaluation to Support the TSTF-505 LAR,” Revision 1 notes that this value was provided in an EPRI letter dated March 11, 2014 (ADAMS Accession No. ML14083A586). The engineering report uses the HCLPF value of 0.25g to calculate a median capacity of 0.67g.

The NRC staff notes that the EPRI letter provides a median capacity, rather than an HCLPF, of 0.25g for Waterford 3. The NRC staff also notes that Safety/Risk Assessment for Generic Issue 199 (ADAMS Accession No. ML100270582), “Implications of Updated Probabilistic Seismic Hazard Estimates in Central and Eastern United States on Existing Plants,” Appendix C, Table C-2, “Plant-Level Fragility Data,” contains an HCLPF value of 0.1g for Waterford 3, which is consistent with the EPRI’s median capacity of 0.25g.

In addition, Engineering Report PSA-WF3-04-01, Tables 2-2 through 2-5 present seismic CDFs (SCDFs) for peak ground acceleration, 10 hertz (Hz), 5 Hz and 1 Hz. The NRC staff notes that these values, ranging from 1.2E-06/yr to 5.5E-06/yr, are very different. This difference is likely caused by using spectral ratios developed from seismic hazard curves that are different from the seismic hazard curves used in this application.

Provide justification for the value of 0.25g for the HCLPF for Waterford 3. If this value cannot be justified, provide an updated value for the HCLPF and make necessary changes to the estimated SCDF in the TSTF-505 LAR.

APLC Question 02 – Calculation of Seismic LERF

As clarified in the SE on TR NEI 06-09, Revision 0-A, other sources of risk (i.e., seismic and other external events) must be quantitatively assessed if they contribute significantly to configuration-specific risk. The SE on TR NEI 06-09, Revision 0-A, also states that bounding analyses or other conservative quantitative evaluations are permitted where realistic PRA models are unavailable.

LAR Enclosure 4, Table E4-1, presents the licensee's bounding analysis of seismic risk to Waterford 3. The LAR describes an adjustment to the conditional large early release probability (CLERP) for internally initiated events to calculate seismic LERF (SLERF) from SCDF. The LAR states that the selected factor of 0.1 is conservative relative to the calculated site risk from other hazards.

The NRC staff notes that the ratio of LERF to CDF for seismic events can be significantly higher than the same ratio for internal events due to the unique nature of seismically induced failures. It is unclear that the selected CLERP of 0.1 represents a conservative or bounding estimate for calculating SLERF in the proposed RICT calculations.

a) Justify that the selected CLERP of 0.1 is conservative for this application. Include the rationale that using a value of 0.1 for the ratio of LERF to CDF for seismic events is conservative or bounding given that internal events random failures do not capture seismically induced failures that may uniquely contribute to LERF.

b) If the approach to estimating SLERF cannot be justified as bounding for this application in response to Item a) above, then provide, with justification, a bounding SLERF for use in RICT calculations.

<u>APLC Question 03 – Seismic Contribution to Incremental Core Damage Probability and Incremental Large Early Release Probability</u>

As clarified in the SE on TR NEI 06-09, Revision 0-A, other sources of risk (i.e., seismic and other external events) must be quantitatively assessed if they contribute significantly to configuration-specific risk. The SE on TR NEI 06-09, Revision 0-A, also states that bounding analyses or other conservative quantitative evaluations are permitted where realistic PRA models are unavailable.

LAR Enclosure 4, Table E4-1, presents the licensee's bounding analysis of seismic risk to Waterford 3. The LAR describes several sources of conservatism inherent in the approach. In Table E4-1, the licensee states, in part:

> The full annual seismic frequency is applied to the seismic 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, the estimated seismic CDF is roughly 12 times the contribution applicable during any RICT.

As CDF and incremental core damage probability (ICDP), or LERF and incremental large early release probability (ILERP), are two different quantities (frequency vs. probability), the meaning of the statement, "the estimated seismic CDF is roughly 12 times the contribution applicable during any RICT" is not clear. Explain the statement quoted above and provide clarification for how SCDF and SLERF are incorporated in the RICT calculations for ICDP and ILERP.

<u>APLC Question 04 – External Hazards Screening</u>

TR NEI 06-09, Revision 0-A, Section 2.3.1, Item 7, states that the impact of other external events risk shall be addressed in the RMTS program. TR NEI 06-09 further states that this may be accomplished via one of three methods. The first method allows the licensee to provide a reasonable technical argument (to be documented prior to implementation of the RMTS program) that the external events that are not modeled in the PRA are not significant contributors to configuration risk. The SE on TR NEI 06-09, Revision 0-A, also states that other external events are treated quantitatively unless it is demonstrated that these risk sources are insignificant contributors to configuration-specific risk.

LAR Enclosure 4, Section 1, provides a list of the external hazards considered in this application. This list was obtained from NUREG-1855, "Guidance on the Treatment of Uncertainties Associated with PRAs in Risk-Informed Decision Making," Volume 1, dated March 2009 (ADAMS Accession No. ML090970525).

NUREG-1855 was subsequently revised in March 2017 (ADAMS Accession No. ML17062A466). The current revision provides a list of hazards in Table 4-1. This comprehensive list is essentially the same list of hazards in nonmandatory Appendix 6-A of ASME/ANS Standard RA-Sa-2009. Therefore, the external hazards considered in this application is only a fraction of the hazards in the PRA standard or NUREG-1855, Revision 1.

In light of the observation above, evaluate the impacts from the external hazards that are listed in the PRA standard or NUREG-1855, Revision 1 but are not considered in LAR Enclosure 4.

Division of Engineering and External Hazards (DEX)
Electrical Engineering Branch (EEEB) Questions

EEEB Question 1 – Supplement Information

Please provide the following:

a) Readable one-line diagrams for alternating current (AC) and direct current (DC) systems including AC and DC buses supplying static uninterruptible power supply (SUPS).

b) Relevant drawings and written material pertaining to diesel generators supporting systems, which indicate their divisional independence.

c) Normal and emergency lineups for AC and DC buses including offsite sources.

d) List of any loads shared between the following:

    i. Engineered Safety Features (ESF) AC divisions
    ii. ESF DC divisions
    iii. SUPS channels

EEEB Question 2 – Divisional Independence

Is divisional independence maintained in associated raceways identified in UFSAR 8.3.1.2.19?

EEEB Question 3 – Safety Related Divisions

Please indicate the number of AC and DC safety-related divisions since the updated final safety analysis report (UFSAR) and TSs have conflicting information?

EEEB Question 4 – Licensing Basis

Is the licensing basis of the plant, a joint loss-of-coolant accident or main steam line break outside containment and loss of offsite power based on UFSAR Table 8.3-1?

EEEB Question 5 – Safety-Related and Nonsafety Loads

Are nonsafety loads supplied during accidents and transients, and if so, how are their electrical faults addressed to protect safety-related loads?

EEEB Question 6 – Limiting Conditions

Please provide the loads supplied by 3AB3-S and its purpose in relation to two AC ESF divisions. Is 3AB3-S considered, a fully redundant division?

EEEB Question 7 – Emergency Operations

Does Waterford 3 use temporary diesel generators during emergency operations?

EEEB Question 8 – Design Success Criteria

For Table E1-1 in the LAR, what is meant by design success criteria for TS LCO 3.8.2.1 of “one battery or one charger supplying DC power to 125 V [volts] DC [VDC] bus,” and which TS required action(s) is this applicable to, 1.a. or 1.b. or both?

For TS 3.8.3, Conditions 1.d., 1.e, 1.f., 1.g, 1.h., and 1.i., is the design success criteria stated in LAR for these TS conditions properly captured given there are four actuation channels with each supplied by an inverter from 480 volts alternating current (VAC) motor control center, or if that fails from a 125 VDC bus?

EEEB Question 9 – Instrumentation

Since there are four actuation channels for instrumentation, typically powered by four motor control centers with two of them supplied by each 4.16 kilovolt ESF bus, why does bus 3AB3-S supply additional inverters?

Division of Safety Systems (DSS)
Technical Specifications Branch (STSB) Questions

STSB Question 1 – Proposed TS Changes (Mark-up)

a) The markup for TS 3.4.3.1 has an arrow with no “insert” - Clarify the intent of the arrow.

b) The proposed administrative controls in TS 6.5.19 paragraph e of Attachment 2 include the phrase “this license amendment.” In lieu of the phrase “this license amendment,” discuss whether the phrases “Amendment # xxx” or, as discussed in the TSTF-505 model SE, “this program” would provide more clarity for this paragraph.

STSB Question 2 – TS Variations

a) TS 3.7.4.a, "Ultimate Heat Sink," in Table E1-1 in Enclosure 1 to the LAR states "The action associated with basin level and temperature are not included in the RICT program. Only the fan operability is included." The TS markup does not indicate any differentiation between operability requirements. Clarify how this statement will only apply to the equipment indicated in Table E1-1.

b) TS 3.6.1.3.a.1 "Containment Air Locks (Atmospheric and Dual)," a specific equivalent condition, is not listed in the cross-reference table and appears to be a variation from TSTF-505. Provide a direct crosswalk for the TSTF-505 equivalent TS and identification of any plant specific action as a variation with technical justification. The apparent equivalent action Standard Technical Specification (STS) 3.6.2 in TSTF-505, Action A is excluded from the RICT.

   A comment on STS 3.6.2 from TSTF-505 states that Condition A contains mitigating actions and require the periodic performance of actions and, therefore, is excluded.

c) TS 3.6.1.7.b, "Containment Ventilation System," a specific equivalent condition, is not listed in the cross-reference table and appears to be a variation from TSTF-505. Provide a direct crosswalk for the TSTF-505 equivalent TS and identification of any plant specific action as a variation with technical justification. The similar STS 3.6.3 Conditions E and F are excluded from RICT.

   A comment on STS 3.6.2 from TSTF-505 states: "Conditions E and F are excluded."

d) TS 3.6.3, "Containment Isolation Valves," needs a direct crosswalk for the TSTF-505 equivalent TS and identification of any plant specific action as a variation with technical justification. Waterford Actions "a" and "e" are plant specific conditions not found in TSTF-505. This variation is not identified and justified in the LAR.

e) The markup in the LAR for TS 3.6.3 does not add the wording "following isolation" per TSTF-505 for Waterford 3.6.3 Actions b, c, f, and g. Provide justification for this variation.

f) The variation in TS 3.7.6.1.a "Control Room Emergency Air Filtration System," from TSTF-505 was not identified in the LAR. Additional technical justification is required for including this Action in the RICT program.

g) The variation in TS 3.7.6.3.a "Control Room Air Temperature - Operating" from TSTF-505 was not identified in the application. Additional technical justification is required for including this Action in the RICT program.

STSB Question 3 - LAR Table E1-1

a) In Table E-1-1 of Enclosure 1 to the LAR, each row should represent a separate TS Action (each proposed application of the RICT Program) and needs to have a clear description of the TS Condition (e.g. one essential services chilled water loop OPERABLE.), and a clear description of the design success criteria (e.g. One of two essential chilled water (ECW) trains operating to provide chilled water to safety-related AHUs.). The design success criteria column should contain the minimum equipment necessary to meet the system function given the condition (i.e. inoperable equipment). (For example, see Shearon Harris Nuclear Power Plant, Unit 1, TSTF-505 LAR (ADAMS Accession No. ML19280C844)).

b) TS LCO Conditions 3.8.1.1.e and 3.8.1.1.c were not included in Table E1-1 of Enclosure 1 to the LAR. Revise the table to include these TS LCO Conditions.

STSB Question 4 - Waterford 3 to Standard Technical Specification Cross Reference Table

a) The cross-reference table should show what is adopted or not in TSTF-505, Revision 2. Every plant specific variation should be identified and have justification in the LAR somewhere (Technical Evaluation section, Table E1-1, cross reference table). TSs 3.6.1.3.a, 3.6.1.7.b (specific STS 3.6.3 equivalent condition), 3.6.3.a through g, 3.7.4..e, 3.7.6.1.a, and 3.7.6.3.a were not included in the cross-reference table, but were included in the markup and Table E1-1. Provide a revised cross reference table to include TSs 3.6.1.3.a, 3.6.1.7.b, 3.6.3.a through g, 3.7.4.e, 3.7.6.1.a, and 3.7.6.3.a.

DSS Nuclear Systems Performance (SNSB) Questions

SNSB Question 1 – Section 3.4.3.1 Statement

In Section 3.4.3.1 of the LAR, “Pressurizer Heaters,”, it states, in part:

> A surrogate event of a stuck open Atmospheric Dump Valve (ADV) will be used as a PRA surrogate for this TS function for RICT evaluations. This represents a conservative and bounding PRA treatment for the loss of the heaters. A stuck open ADV would (like failed heaters) limit the ability of the system to maintain RCS pressure as desired. This failure also has secondary side impacts which ensures that the PRA related risk results would be bounding for the loss of Pressurizer Heater function for RICT considerations.

Provide analyses results, test data or other acceptable documents to justify the above statement.

DSS Containment and Plant Systems Branch (SCPB) Questions

SCPB Question 1 – Proposed Changes to TS 3.6.3

The wording of proposed TS 3.6.3, Action e., f., or g. for penetrations not associated with closed systems does not appear to unambiguously ensure an operable containment isolation valve will be present in each containment penetration flow path, and, therefore, the action may not ensure maintenance of the containment isolation function during the proposed RICT. The statement prior to Action e. specifies that at least one isolation valve be maintained operable in each penetration that is open. However, this statement does not require an operable isolation valve in each containment penetration flow path, as specified in standard TS 3.6.3, Condition A, cited in TSTF-505. Therefore, the NRC staff notes that the action could be entered under a condition representing a loss of containment function because penetrations with more than one flow path may not have an operable barrier in each flow path. Provide justification of the TS 3.6.3 statement as ensuring each penetration flow path would have an operable isolation valve or clarify the Action statement to limit the applicability of the proposed RICTs to conditions that would not result in a loss of containment function.

SCPB Question 2 – Proposed Changes to TS 3.6.1.7.b

The wording of proposed TS Action 3.6.1.7.b appears ambiguous because it does not clearly limit the number of inoperable valves to only one of the two purge valves for the purge exhaust penetration. The NRC staff considered the action ambiguous because it begins with the statement "With a containment purge supply and/or exhaust isolation valve(s) having a measured leakage rate exceeding the limits...", and the staff determined that two exhaust isolation valves with excessive leakage could satisfy this condition statement. Therefore, the staff found that the action could be entered under a condition representing a loss of containment function because redundant exhaust valves in one penetration could have excessive leakage. Provide justification of the statement as applying only when no more than one valve per penetration has excessive leakage, or clarify the Action statement to limit the applicability of the proposed risk-informed completion times to conditions that would not result in a loss of containment function.

SCPB Question 3 – Proposed Changes to TS 3.7.1.2.d

Proposed changes to TS Action 3.7.1.2.d to apply an RICT to conditions where the emergency feedwater (EFW) system is inoperable for reasons other than those conditions described in Actions a., b., and c., which involve inoperability of one of two steam supplies to the turbine-driven EFW pump combined with zero, one, and two motor-driven EFW pumps, respectively. In addition, TS Action 3.7.1.2.d. states that the EFW system is able to deliver at least 100 percent flow to either SG. The bases for TS 3.7.1.2 (ADAMS Accession No. ML020660508) address conditions involving inoperability of the turbine-driven EFW pump, inoperability of both motor-driven EFW pumps, or inoperability of one of two redundant flow paths. Table E1-2, "In Scope TS/LCO Conditions RICT Estimate," in Enclosure 1 to the LAR indicated an estimated RICT for this action not involving inoperable pumps. Provide clarification of the condition assumed for evaluation of the 10-day estimated RICT for Action d. (not pump related) and how the estimated completion times would change for conditions involving inoperability of the turbine-driven EFW pump alone for reasons other than one inoperable steam supply and, separately, inoperability of both motor driven EFW pumps.

SCPB Question 4 – Proposed Changes to TS 3.7.4.e

The proposed TS changes provided in Attachment 2 to the LAR indicate addition of the RICT provision to TS Action 3.7.4.e, which applies when either or both wet cooling tower basin cross-connect valves are not operable for makeup. However, Tables E1-1 and E1-2 in Enclosure 1, and Enclosure 13, "Waterford 3 to Standard Technical Specification Cross Reference," do not address that condition. Also. the specific proposed changes are not listed in Attachment 1, "Description and Assessment of the Proposed Change," to the LAR. Request clarification if TS Action 3.7.4.e was intended to be within the scope of the proposed amendment request and provide appropriate supporting information if Action e. was intended to be within the LAR scope.

SCPB Question 5 – Proposed Changes to TS 3.7.6.1

For the proposed change to TS 3.7.6.1, Action a. is related to the control room emergency air filtration system. The filtration system provides control room isolation for protection from toxic gases and smoke, smoke purge for fires in the control room vicinity, and filtration of outside air during radiological releases from design basis events. Table E1-1 in Enclosure 1, includes the following note: "All the components in the MCR [main control room] HVAC [heating, ventilation, and air conditioning]/filtration system are not in the PRA model, but several Air Handling Units (AHUs), fans, and dampers are included allowing for a bounding evaluation using the CRMP." However, Enclosure 4, Table E4-1, indicated that toxic gas events were excluded from the PRA model supporting the CRMP based on low frequency of occurrence. Although the smoke removal function may be bounded in the fire PRA, the radiological protection function does not appear to be adequately represented by consideration of core damage and large early release frequencies. Please describe the initiating events and functions performed by the main control room emergency filtration system that are modeled in the CRMP tool and how the RICT reflects the importance of the functions of the control room emergency air filtration system analyzed in the safety analysis report, consistent with 10 CFR 50.36(b).

SUBJECT: WATERFORD STEAM ELECTRIC STATION, UNIT 3 – REGULATORY AUDIT IN SUPPORT OF LICENSE AMENDMENT REQUEST TO REVISE TECHNICAL SPECIFICATIONS TO ADOPT RISK-INFORMED COMPLETION TIMES (EPID L-2021-LLA-0014) DATED OCTOBER 7, 2021

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