From: Robert Kuntz
Sent: Wednesday, March 6, 2024 6:40 AM
To: rebecca.steinman@constellation.com
Subject: Request for Additional Information RE: TSTF-505 and 10 CFR 50.69 license amendments
Attachments: Quad Cities TSTF505 and 50.69 RAI.docx

Ms. Steinman,

By letters dated June 8, 2023 (Agencywide Documents Access and Management System (ADAMS) Accession Nos. ML23159A249 and ML23159A253, respectively), Constellation Energy Generation, LLC (Constellation, the licensee) submitted two license amendment requests (LARs) for Quad Cities Nuclear Power Station (Quad Cities), Units 1 and 2. The proposed amendments would modify Renewed License Nos. DPR-29 and DPR-30, and the Technical Specifications (TSs) to adopt Technical Specifications Task Force (TSTF) Traveler TSTF-505, Revision 2, "Provide Risk-Informed Extended Completion Times, RITSTF [Risk-Informed Technical Specification Task Force] Initiative 4b" (ML18183A493), and to allow for the implementation of the provisions of Title 10 of the *Code of Federal Regulations*, Part 50 (10 CFR 50), section 50.69, "Risk-informed categorization and treatment of structures, systems, and components [SSCs] for nuclear power reactors."

The NRC staff has determined that additional information is needed to support its review. The attached is the NRC staff's request for additional information (RAI). During clarification calls held on February 29 and March 5, 2024, a 30-day response was agreed upon. Therefore, the NRC staff expects a response to the attached RAI by April 5, 2024.

In addition, based on the discussions during the clarification calls, the staff has revised the following items to provide clarification on the information sought by the NRC staff:

APLA RAI 02, Item (d) was revised from:

Additionally, the Quad Cities, Units 1 and 3, PRA does not state a failure probability for post-injection containment failure. This failure probability may impact the conservatism of the probability of large dry containment failure. Confirm whether post-injection containment failure mode was specifically analyzed in Appendix C.

to:

Additionally, the Quad Cities, Units 1 and 2, PRA does not model post-containment failure injection. Modeling of post-containment failure injection may impact the conservatism of the stated probability of 6E-02 for large drywell failure. Explain how post-containment failure injection was specifically analyzed, and confirm whether this failure mode is bounded by the probability of drywell failure.

APLA RAI 04 item (c) was revised from:

Explain how the Quad Cities, Units 1 and 2, PRA and PARAGON CRMP tools account for seasonal variations in the failure probabilities of modeled components.

to:

Explain how the Quad Cities, Units 1 and 2, PRA accounts for seasonal variations in the modeling of affected components, and how these variations, if any, are reflected in the PARAGON CRMP (RTR) tool.

Sincerely,

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Hearing Identifier: NRR_DRMA
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DRAFT REQUEST FOR ADDITOINAL INFORMATION

LICENSE AMENDMENT REQUESTS TO REVISE TECHNICAL SPECIFICATIONS TO

ADOPT TSTF-505, REVISION 2 AND IMPLEMENT 10 CFR 50.69

CONSTELLATION ENERGY GENERATION, LLC

QUAD CITIES NUCLEAR POWER STATION, UNITS 1 AND 2

DOCKET NOS. 50-254 AND 50-265

By letters dated June 8, 2023 (Agencywide Documents Access and Management System (ADAMS) Accession Nos. ML23159A249 and ML23159A253, respectively), Constellation Energy Generation, LLC (Constellation, the licensee) submitted two license amendment requests (LARs) for Quad Cities Nuclear Power Station (Quad Cities), Units 1 and 2. The proposed amendments would modify Renewed License Nos. DPR-29 and DPR-30, and the Technical Specifications (TSs) to adopt Technical Specifications Task Force (TSTF) Traveler TSTF-505, Revision 2, "Provide Risk-Informed Extended Completion Times, RITSTF [Risk-Informed Technical Specification Task Force] Initiative 4b" (ML18183A493), and to allow for the implementation of the provisions of Title 10 of the *Code of Federal Regulations*, Part 50 (10 CFR 50), section 50.69, "Risk-informed categorization and treatment of structures, systems, and components [SSCs] for nuclear power reactors."

The NRC staff has determined that additional information is needed to support its review. The following is the NRC staff's draft request for additional information.

APLA RAI 01

Concerning the quality of the PRA model, Nuclear Energy Institute (NEI) 06-09-A, "Risk-Informed Technical Specifications Initiative 4b Risk-Managed Technical Specifications (RMTS) Guidelines," Revision 0-A (ML12286A322), states that Regulatory Guide (RG) 1.174, "An Approach for Using Probabilistic Risk Assessment in Risk-Informed Decisions on Plant-Specific Changes to the Licensing Basis" (ML17317A256) and RG 1.200, "Acceptability of Probabilistic Risk Assessment Results for Risk-Informed Activities" (ML090410014) 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.

Regarding digital instrument and control (I&C), the NRC staff notes the lack of consensus industry guidance for modeling these systems in plant PRAs to be used to support risk-informed regulatory applications. In addition, 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, although 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 risk informed completion time (RICT) and 50.69 programs.

Table E9-1 of Enclosure 9 to the RICT (TSTF-505) LAR and the table in Attachment 6 of the 10 CFR 50.69 LAR identifies the digital feedwater control (DFWC) failure probabilities as a potential key uncertainty and performed a sensitivity of increasing the likelihood that the DFWC would result in overfilling the reactor pressure vessel (RPV). The NRC staff notes that another failure mode consideration is loss of feedwater to the RPV. It is unclear to the NRC staff if the sensitivity study failure mode is bounding for this uncertainty. Therefore, address the following:

a) Provide justification that the DFWC failure mode addressed in the LAR sensitivity is the bounding failure mode of this control system.

b) If another failure mode is determined to be bounding, provide justification (e.g., describe and provide the results of a sensitivity study) that demonstrates the modeling uncertainty associated with crediting digital I&C systems has an inconsequential impact on the RICT calculations and 50.69 categorization.

Clarify whether digital I&C systems, other than DFWC, are credited in the PRA models that will be used in the RICT and 50.69 programs.

c) If other digital I&C systems are credited in the PRA models and will be used in the RICT or 50.69 programs, provide justification (e.g., describe and provide the results of a sensitivity study) that demonstrates the modeling uncertainty associated with crediting digital I&C systems has an inconsequential impact on the RICT calculations and 50.69 categorizations.

   Alternatively, for RICT, if a justification is not provided, identify which LCOs are determined to be impacted by digital I&C systems modeling for which risk management actions (RMAs) will be applied during a RICT. Explain and justify the criteria used to determine what level of impact to the RICT calculation requires additional RMAs.

APLA RAI 02

The NRC staff safety evaluation to NEI 06-09, Revision 0, specifies that the LAR should identify key assumptions and sources of uncertainty and to assess/disposition each as to their impact on the RMTS application. LAR Enclosure 9, Table E9-1 identifies the key assumptions and sources of uncertainty for the internal events and fire PRA models and provides dispositions for each source of uncertainty for this TSTF-505 application. NRC staff reviewed these dispositions and is unclear that some uncertainty dispositions fully addressed the potential impact to RICT calculations. Therefore, address the following:

TSTF-505 LAR Enclosure 9, Table E9-1 regarding core cooling success following containment failure, states that the sensitivity study demonstrates that all risk metrics are sensitive to this uncertainty. NRC review of the TSTF-505 LAR, Section 8.1 of the Assessment of Key Assumptions and Sources of Uncertainty Notebook demonstrates impacts on CDF and LERF ranging from thirty-three to eighty-one percent. It appears that this source of uncertainty could plausibly impact this application. However, the TSTSF-505 LAR reasons that the increased factor value is not considered credible. The NRC staff notes that it requires insights from credible sensitivity studies for its review process and determination. It is unclear to the NRC staff that this assumption has no impact on the RICT or 50.69 programs. Therefore, address the following:

a) Clarify what increased factor value should be used for this sensitivity. Include in this discussion justification that the selected increased factor value appropriately bounds the increase in risk associated with this uncertainty.

b) Based on the response to part (a) above, provide justification that the uncertainty associated with core cooling success following containment failure does not significantly impact any RICT calculation or 50.69 categorizations.

c) Alternatively to part (b) above, for RICT, provide how this source of uncertainty, such as additional risk management actions (RMAs), would be addressed in the RICT program.

d) Additionally, the Quad Cities, Units 1 and 2, PRA does not model post-containment failure injection. Modeling of post-containment failure injection may impact the conservatism of the stated probability of 6E-02 for large drywell failure. Explain how post-containment failure injection was specifically analyzed, and confirm whether this failure mode is bounded by the probability of drywell failure.

<u>APLA RAI 03</u>

RG 1.200, Revision 2, states, in part: "The base PRA serves as the foundational representation of the as-built and as-operated plant necessary to support an application."

The TSTF-505 LAR does mention the existence of interconnected auxiliary systems between units. The NRC staff notes that for some of these systems, it appears the sharing of a system is not consistent between units. Enclosure 8 to the TSTF-505 LAR states that the Real Time Risk (RTR) model can represent the availability of these shared components. However, it is unclear to the NRC staff if all operational aspects, such as alternate alignments, were excluded from the PRA models. In addition, multi-unit events (e.g., loss of offsite power and seismic events), credit for a shared system may be limited to one unit.

Clarify what systems are shared, how they are shared, and whether they can support the other unit in an accident. Explain how the shared systems are credited for each unit in the PRA models. This discussion should also address the following:

a) Explain whether shared systems are credited in the internal events, including flood and fire PRA models for both units and, if so, identify those systems.

b) If shared systems are credited in the RTR model that supports the RICT calculations, then explain how the shared system is modelled for each unit in a dual unit event demonstrating that shared systems are not over-credited in the PRA models.

c) If a shared system is credited in the RTR model that supports the RICT calculations and the impact of events that can create a concurrent demand for the system shared by both units is not addressed in the PRA models, then justify that this exclusion has an inconsequential impact the RICT calculations.

d) Explain how the PARAGON CRMP model displays the availability of shared systems on the operator screen for both units. Confirm that the PARAGON tool considers the unavailability of a shared system for both units.

<u>APLA RAI 04</u>

The Tier 3 assessment in RG 1.177, "An Approach for Plant-specific, Risk-informed Decision-making: Technical Specifications," Revision 2 (ML20164A034), stipulates that a licensee should develop a program that ensures the risk impact of out-of-service equipment is appropriately evaluated prior to performing any maintenance activity. NEI 06-09-A and its associated NRC SE (ML071200238) state that, for the impact of seasonal changes, either conservative assumptions should be made, or the PRA should be "adjusted appropriately to reflect the current (e.g., seasonal or time of cycle) configuration."

Enclosure 8 of the TSTF-505 LAR mentions modifications related to the impact of outside temperatures for one structures, systems, and component (SSC). However, it does not appear to state if other modeling adjustments are needed to account for seasonal and time of cycle dependencies and what kind of adjustments will be made. Therefore, address the following to clarify the treatment of seasonal and time of cycle variations:

a) Explain how the RICT calculations address changes in PRA data points, basic events, and SSC operability constraints as a result of extreme weather conditions, seasonal variations, other environmental factors, or time of cycle. Also, explain how these adjustments are made in the configuration risk management program (CRMP) model and how this approach is consistent with the guidance in NEI 06-09-A and its associated NRC final SE.

b) Describe the criteria used to determine when PRA adjustments due to extreme weather conditions, seasonal variations, other environmental factors, or time of cycle variations need to be made in the CRMP model and what mechanism initiates these changes.

c) Explain how the Quad Cities, Units 1 and 2, PRA accounts for seasonal variations in the modeling of affected components, and how these variations, if any, are reflected in the PARAGON CRMP (RTR) tool.

<u>APLA RAI 05</u>

The NRC SE for NEI 06-09-A, states in part: "The impact of the proposed change should be monitored using performance measurement strategies." NEI 06-09-A considers the use of NUMARC 93-01, Revision 4F, "Industry Guideline for Monitoring the Effectiveness of Maintenance at Nuclear Power Plants" (ML18120A069), as endorsed by RG 1.160, "Monitoring the Effectiveness of Maintenance at Nuclear Power Plants," Revision 4 (ML18220B281), for the implementation of the Maintenance Rule. NUMARC 93-01, Section 9.0, contains guidance for the establishment of performance criteria.

In addition, the NEI 06-09-A methodology satisfies the five key safety principles specified in RG 1.177, Revision 2 relative to the risk impact due to the application of a RICT. Moreover, NRC staff position C.3.2 provided in RG 1.177, Revision 2, for meeting the fifth key safety principle acknowledges the use of performance criteria to assess degradation of operational safety over a period. It is unclear how the licensee's RICT program captures performance monitoring for the SSCs within the scope of the RMTS program. Therefore:

a) Confirm that the Quad Cities, Units 1 and2, Maintenance Rule program incorporates the use of performance criteria to evaluate SSC performance as described in NUMARC 93-01, as endorsed by RG 1.160.

b) Alternatively, describe the approach or method used for SSC performance monitoring, as described in NRC staff position C.3.2 of RG 1.177, Revision 2, for meeting the fifth key safety principle. In the description, include criteria (e.g., qualitative, or quantitative) along with the appropriate risk metrics and explain how the approach and criteria demonstrate the intent to monitor the potential degradation of SSCs in accordance with the NRC SE for NEI 06-09-A.

APLA RAI 06

The NRC's safety evaluation for NEI 06-09-A specifies that the LAR should provide a comparison of the TS functions to the PRA modeled functions to show that the PRA modeling is consistent with the licensing basis assumptions or to provide a basis for when there is a difference. Table E1-1 of the TSTF-505 LAR, Enclosure 1, identifies each Limiting Condition for Operation (LCO) in the TSs proposed for inclusion in the RICT program. The table also describes whether the systems and components covered by the LCO are modeled in the PRA and, if so, presents both the design success criteria and PRA success criteria. For certain LCOs, the table explains that the associated SSCs are not modeled in the PRAs but will be represented using a surrogate event that fails the function performed by the SSC. For some LCOs, the LAR did not provide an adequate description for the NRC staff to conclude that the PRA modeling will be sufficient.

a) Regarding TS LCO 3.3.5.1.B, Table E1-1 states that, for emergency core cooling system (ECCS) actuation instrumentation for core spray (CS), low pressure coolant injection (LPCI), high pressure coolant injection (HPCI), and diesel generators (DGs) that SSCs associated with Functions 3.a and 3.b are not explicitly modeled in the PRA. However, the associated Table E1-1 comments entry states that all of the SSCs can be explicitly modeled in the RTR tool. The NRC staff notes that the Functions 3.a and 3.b appear to be associated with HPCI instrumentation. It is unclear to the staff what system is associated with Functions 3.a and 3.b and if this associated instrumentation is incorporated into the RTR tool.

    i. Clarify the system(s) associated with TS LCO 3.3.5.1.B Functions 3.a and 3.b. Include in this response if the actuation instrumentation related to these two functions are explicitly modeled in the RTR tool.

    ii. If the actuation instrumentation related to Functions 3.a and 3.b of TS LCO 3.3.5.1.B are not explicitly modeled in the RTR tool, then provide the following information:

        a. Identify the RTR model surrogates to be used for Functions 3.a and 3.b of TS LCO 3.3.5.1.B

        b. Provide justification that the surrogate(s) is related and bounds both Functions 3.a and 3.b of TS LCO 3.3.5.1.B.

b) Regarding TS LCO 3.3.5.1.C, Table E1-1 states that, for ECCS actuation instrumentation for CS, LPCI, and DGs that SSCs associated with Functions 3.c and 3.g are not explicitly modeled in the PRA. However, the associated Table E1-1 comments entry states that all of the SSCs can be explicitly modeled in the RTR tool. It is unclear to the staff what system is associated with Functions 3.c and 3.g and if the associated instrumentation is in the RTR tool.

i. Clarify the system(s) associated with TS LCO 3.3.5.1.C Functions 3.c and 3.g. Include in this response if the actuation instrumentation related to these two functions are explicitly modeled in the RTR tool.

ii. If the actuation instrumentation related to Functions 3.c and 3.g of TS LCO 3.3.5.1.C are not explicitly modeled in the RTR tool, then provide the following information:

   a. Identify the RTR model surrogates to be used for Functions 3.c and 3.g of TS LCO 3.3.5.1.C

   b. Provide justification that the surrogate(s) is related and bounds both Functions 3.c and 3.g of TS LCO 3.3.5.1.C.

c) Regarding TS LCO 3.6.1.2.C, Table E1-1 states that, for primary containment air locks not modeled, that a large pre-existing containment isolation failure that is modeled will be used as a surrogate. It is unclear to the staff how pre-existing containment isolation failure is either conservative or bounding.

   Provide justification that the surrogate conservatively bounds TS LCO 3.6.1.2.C. Explain whether this justification was specifically modeled, or included as an assumption.

d) Regarding TS LCO 3.6.1.3.A, Table E1-1 states that, for primary containment isolation valves not modeled, that a large pre-existing containment isolation failure that is modeled will be used as a surrogate. It is unclear to the staff how pre-existing containment isolation failure is either conservative or bounding.

   Provide justification that the surrogate conservatively bounds TS LCO 3.6.1.3.A.

<u>APLA RAI 07</u>

Section 2 of Enclosure 9 of the RICT LAR states that FLEX is credited in the Quad Cities internal events PRA (FPIE), which includes internal flooding, and the FPRA.

NRC memorandum dated May 6, 2022[1] provides the NRC's staff updated 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[2]. The NRC staff states in Conclusion 4 of the memo: "Licensees that choose not to use the generic failure probabilities in PWROG-18042 to develop plant-specific failure probabilities for portable FLEX equipment modeled in PRAs used for risk-informed applications should submit a justification for the methods and probabilities used to the NRC for review and approval."

It appears that NUREG-6928 fixed equipment failure rates, including a 2x increase, were used as probabilities for FLEX portable equipment. It is unclear to the NRC staff how the Quad Cities, Units 1 and 2, approach satisfies the concerns of Conclusion 4.

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[1] U.S. NRC memorandum, "Updated Assessment of Industry Guidance for Crediting Mitigating Strategies in Risk Assessments," dated May 6, 2022 (ML22014A084).
[2] U.S. NRC, "Acceptability of Probabilistic Risk Assessment Results for Risk-Informed Activities," RG 1.200, Revision 3, December 2020 (ML20238B871).

The enclosure explains how NRC's "Updated Assessment of Industry Guidance for Crediting Mitigating Strategies in Probabilistic Risk Assessments" (ML22014A084) is addressed in the modeling of FLEX.

Address the following:

a) Propose a mechanism to incorporate updated FLEX parameter values in accordance with PWROG-18042-N into the Quad Cities PRA models used for RICT calculations prior to implementing the RMTS program.

   -OR-

   Alternatively, identify the LCO conditions impacted by the treatment of this modelling uncertainty for which RMAs will be applied during a RICT. Include discussion of the kinds of RMAs that would be applied and justification that the RMAs will be sufficient to address the modeling uncertainty.

b) Provide a discussion detailing the methodology used to assess operator actions related to installation and operation of FLEX equipment. The discussion should include:

   i. A list of the FLEX-related operator actions and a summary description of the plant-specific HRA used as the basis to develop the HEPs for each operator action. Include an evaluation of the HRA associated with declaration of Extended Loss of AC Power (ELAP).

   ii. If the FLEX-related HRA is not in accordance with the NRC memorandum dated May 6, 2022, justification that the HRA assumptions have an inconsequential impact on the RICT calculations.

   iii. If, in response to part ii~~i)~~ above, it cannot be determined that the cited assumptions have an inconsequential impact on the estimated RICTs, then identify the LCO conditions impacted by the treatment of this modelling uncertainty for which RMAs will be applied during a RICT. Include a discussion of the programmatic changes that the licensee will consider in order to compensate for this uncertainty and the basis for their consideration (e.g., identification of additional RMAs and justification that they are sufficient to address the modeling uncertainty).

c) If the PRA modeling of FLEX equipment and/or operator actions is revised or updated to be in accordance with the NRC memorandum dated May 6, 2022, provide justification that the revisions do not meet the definition of an PRA upgrade as defined by RG 1.200.

   -OR-

   Alternatively, if justification cannot be provided, propose a mechanism to conduct a focused-scope peer review (FSPR) regarding incorporation of the PWR Owners Group FLEX equipment reliability modeling and/or EPRI FLEX HRA methodology for the ANO-2 PRA models. Include in the mechanism to close out all Facts and Observations (F&Os) that result from the FSPR prior to implementing the RMTS program.

APLB RAI 01

The TSTF-505 LAR, Enclosure 2, provides the history of the Fire PRA (FPRA) peer review but does not appear to discuss methods used in the FPRA. Methods may have been used in the FPRA that deviate from guidance in NUREG/CR-6850, "EPRI/NRC-RES Fire PRA Methodology for Nuclear Power Facilities," (ML052580075, ML052580118, and ML103090242), or other acceptable guidance (e.g., frequently asked questions (FAQs), NUREGs, or interim guidance documents).

a) Identify methods used in the FPRA that deviate from guidance in NUREG/CR-6850 or other acceptable guidance.

b) If such deviations exist, then justify their use in the FPRA and impact on the RICT program.

c) As an alternative to item b above, add an implementation item to replace those methods with a method acceptable to NRC prior to the implementation of the RICT program. Include a description of the replacement method along with justification that it is consistent with NRC accepted guidance.

APLB RAI 02

The key factors used to justify using transient fire reduced heat release rates (HRRs) below those prescribed in NUREG/CR-6850 are discussed in the June 21, 2012, letter from Joseph Giitter, U.S. Nuclear Regulatory Commission, to Biff Bradley, NEI, "Recent Fire PRA Methods Review Panel Decisions and Electrical Power Research Institute (EPRI) 1022993, 'Evaluation of Peak Heat Release Rates in Electrical Cabinet Fires'." (ML12172A406).

If any reduced transient HRRs below the bounding 98% HRR of 317 kW from NUREG/CR-6850 were used, discuss the key factors used to justify the reduced HRRs. Include in this discussion:

a) Identification of the fire areas where a reduced transient fire HRR is credited and what reduced HRR value was applied.

b) A description for each location where a reduced HRR is credited, and a description of the administrative controls that justify the reduced HRR including how location-specific attributes and considerations are addressed. Include a discussion of the required controls for ignition sources in these locations and the types and quantities of combustible materials needed to perform maintenance. Also, include discussion of the personnel traffic that would be expected through each location.

c) The results of a review of records related to compliance with the transient combustible and hot work controls.

APLB RAI 03

FAQ 13-0004, "Clarifications on Treatment of Sensitive Electronics" (ML13322A085) provides supplemental guidance for application of the damage criteria provided in Sections 8.5.1.2 and H.2 of NUREG/CR-6850, Volume 2, for solid-state and sensitive electronics.

a) Describe the treatment of sensitive electronics for the FPRA and explain whether it is consistent with the guidance in FAQ 13-0004, including the caveats about configurations that can invalidate the approach (i.e., sensitive electronics mounted on the surface of cabinets and the presence of louver or vents).

b) If the approach cannot be justified to be consistent with FAQ 13-0004, then justify that the treatment of sensitive electronics has no impact on the RICT calculations.

c) As an alternative to item b above, add an implementation item to replace the current approach with an acceptable approach prior to the implementation of the RICT program. Include a description of the replacement method along with justification that it is consistent with NRC accepted guidance.

APLB RAI 04

NUREG-1921, "EPRI/NRC-RES Fire Human Reliability Analysis Guidelines- Final Report," (ML12216A104), discusses the need to consider a minimum value for the joint probability of multiple human failure events (HFEs) in human reliability analyses (HRAs).

NUREG-1921 refers to Table 2-1 of NUREG-1792, "Good Practices for Implementing Human Reliability Analysis (HRA)," (ML051160213), which recommends that joint human error probability (HEP) values should not be below 1E-5. Table 4-4 of EPRI 1021081, "Establishing Minimum Acceptable Values for Probabilities of Human Failure Events," provides a lower limiting value of 1E-6 for sequences with a very low level of dependence. Therefore, the guidance in NUREG-1921 allows for assigning joint HEPs that are less than 1E-5, but only through assigning proper levels of dependency.

The TSTF-505 LAR, Enclosure 9, Table E9-1 in regarding joint human error probabilities (JHEPs) in the FPRA, states that a sensitivity study was performed since the analysis implements a JHEP floor of 1E-06. The sensitivity study used the JHEP floor value of 1E-05, which is consistent with industry guidance. The LAR states that the sensitivity demonstrated that this source of uncertainty had a slight impact on FPRA results. However, during the staff's portal review of the Quad Cities, Units 1 and 2, FPRA Sensitivity Analysis Notebook, it was noted that the sensitivity results provided in Section 4.2.3 demonstrates a 9 percent impact in overall increase in core damage frequency (CDF) and large early release frequency (LERF) for Quad Cities, Unit 1. In addition, Section 8.8 of the Assessment of Key Assumptions and Sources of Uncertainty Notebook demonstrates a fourteen percent impact on TS LCO 3.8.7.C.4 RICT calculation and provides results for only seven other TS LCO proposed for the RICT program. It is unclear to the NRC staff that this assumption has no impact on the RICT program. Therefore, address the following:

a) Provide justification, such as a RICT sensitivity study that the that the minimum joint HEP value has no impact on the remaining TS LCOs proposed for the RICT application.

b) If, in response part (a), if it cannot be justified that the minimum joint HEP value has no impact on the application, then provide the following:

   i. Confirm that each joint HEP value used in the FPRA below 1E-5 includes its own justification that demonstrates the inapplicability of the NUREG-1792 lower value guideline (i.e., using such criteria as the dependency factors identified in NUREG-1921 to assess level of dependence). Provide an estimate of the number of these joint HEP values below 1.0E-5, discuss the range of values, and provide at least two different examples where this justification is applied.

ii. If joint HEP values used in the FPRA below 1E-5 cannot be justified, add an implementation item to set these joint HEPs to 1E-5 in the FPRA prior to the implementation of the RICT program.

APLB RAI 05

NUREG-2178, Volume 1 "Refining and Characterizing Heat Release Rates from Electrical Enclosures During Fire (RACHELLE-FIRE), Volume 1: Peak Heat Release Rates and Effect of Obstructed Plume," (ML16110A14016) contains refined peak HRRs, compared to those presented in NUREG/CR-6850, and guidance on modeling the effect of plume obstruction. Additionally, NUREG-2178 provides guidance that indicates that the obstructed plume model is not applicable to cabinets in which the fire is assumed to be located at elevations of less than one-half of the cabinet.

a) If obstructed plume modeling was used, then indicate whether the base of the fire was assumed to be located at an elevation of less than one-half of the cabinet.

b) Justify any modelling in which the base of an obstructed plume is located at less than one half of the cabinet's height.

c) As an alternative to item b above, add an implementation item to remove credit for the obstructed plume model in the FPRA prior to the implementation of the RICT program.

APLB RAI 06

Guidance in FAQ 08-0042 from Supplement 1 of NUREG/CR-6850 applies to electrical cabinets below 440V. With respect to Bin 15 as discussed in Chapter 6, it clarifies the meaning of "robustly or well-sealed." Thus, for cabinets of 440V or less, fires from well-sealed cabinets do not propagate outside the cabinet. For cabinets of 440V and higher, the original guidance in Chapter 6 remains and states that Bin 15 panels which house circuit voltages of 440V or greater are counted because an arcing fault could compromise panel integrity (an arcing fault could burn through the panel sides, but this should not be confused with the high energy arcing fault type fires)." FPRA FAQ 14-0009, "Treatment of Well-Sealed MCC Electrical Panels Greater than 440V" (ADAMS Accession No. ML15119A176) provides the technique for evaluating fire damage from MCC cabinets having a voltage greater than 440V. The NRC guidance recommends that propagation of fire outside the ignition source panel should be evaluated for all MCC cabinets that house circuits of 440V or greater.

a) Describe how fire propagation outside of well-sealed MCC cabinets greater than 440V is evaluated.

b) If well-sealed cabinets less than 440V are included in the Bin 15 count of ignition sources, provide justification for using this approach as this is contrary to the guidance.

APLB RAI 07

NUREG/CR-6850, Section 6, "Fire Ignition Frequencies," and FAQ 12-0064 "Hot Work/Transient Fire Frequency Influence Factors" (ADAMS Accession No. ML12346A488) describe the process for assigning influence factors for hot work and transient fires. Provide the following regarding application of this guidance:

a) Indicate whether the methodology used to calculate hot work and transient fire frequencies applies influencing factors using NUREG/CR-6850 guidance or FAQ 12-0064 guidance.

b) Indicate whether administrative controls are used to reduce transient fire frequency, and if so, describe and justify these controls.

c) Indicate whether you have any combustible control violations and discuss your treatment of these violations for the assignment of transient fire frequency influence factors. For those cases where you have violations and have assigned an influence factor of 1 (Low) or less, indicate the value of the influence factors you have assigned and provide your justification.

d) If you have assigned an influencing factor of "0" to Maintenance, Occupancy, or Storage, or Hot Work for any fire physical analysis units (PAUs) provide justification.

e) If a weighting factor of "50" was not used in any fire PAU, provide a sensitivity study that assigns weighting factors of "50" per the guidance in FAQ 12-0064.

<u>APLB RAI 08</u>

Quad Cities, Units 1 and 2 are adjoined and thus have common areas. The risk contribution from fires originating in one unit must be addressed for impacts to the other unit given the physical proximity of the other unit, common areas, and existence of shared systems. Therefore, address the following:

a) Explain how the risk contribution of fires originating in one unit is addressed for the other unit given impacts due to the physical proximity of equipment and cables in one unit to equipment and cables in the other unit. Include identification of locations where fire in one unit can affect components in the other unit and explain how the risk contributions of such scenarios are allocated in the LAR.

b) Explain how the contributions of fires in common areas are addressed, including the risk contribution of fires that can impact components in both units.

c) Explain the extent to which systems are shared by both units and whether shared systems are credited in the PRA models for both units. If shared systems are credited in the PRA models for each unit, then explain how the PRAs address the possibility that a shared system is demanded in both units in response to a single initiating event or fire initiator.

<u>APLB RAI 09</u>

Traditionally, the cabinets on front face of the main control board (MCB) have been referred to as the MCB for purposes of FPRA. Appendix L of NUREG/CR-6850, (ML052580075) provides a refined approach for developing and evaluating those fire scenarios. FPRA FAQ 14-0008, “Main Control Board Treatment” dated July 22, 2014 (ML14190B307) clarifies the definition of the

MCB and provides guidance for when to include the cabinets on the back side of the MCB as part of the MCB for FPRA. It is important to distinguish between MCB and non-MCB cabinets because misinterpretation of the configuration of these cabinets can lead to incomplete or incorrect fire scenario development. This FAQ also provides several alternatives to NUREG/CR-6850 for using Appendix L to treat partitions in an MCB enclosure. Therefore, address the following:

a) Describe the main control room MCB configuration, and use the guidance in FAQ 14-0008, to determine whether cabinets on the rear side of the MCB are a part of the MCB. Provide justification using the FAQ guidance.

b) If the cabinets on the rear side of the MCB are part of a single integral MCB enclosure using the definition in FAQ 14-0008 confirm that guidance in FAQ 14-0008 was used to develop fire scenarios in the MCB and determine the frequency of those scenarios.

c) If the cabinets on the rear side of the MCB are part of a single integral MCB enclosure and the guidance in FAQ 14-0008 was not used to develop fire scenarios involving the MCB provide a description of how the fire scenarios for the backside cabinets are developed and an explanation of how the treatment aligns with NRC accepted guidance.

d) If in response to parts (c) above, the current treatment of the MCB and those cabinets on the rear side of the MCB cannot be justified using NRC accepted guidance, then justify that the treatment has no impact on the RICT calculations. Alternatively, propose a mechanism that ensures that the FPRA is updated to treat the MCB enclosure consistent with the guidance in FAQ 14-0008, prior to implementation of the RICT program.

APLB RAI 10

The TSTF-505 LAR states in part that the Internal FPRA model was developed consistent with NUREG/CR-6850 and only utilizes NRC approved methods. As part of the ongoing PRA maintenance and update process described in the LAR, the licensee will address Internal FPRA methods approved by the NRC since the development of the Internal FPRA. Furthermore, the TSTF-505 LAR specifies that a full-scope FPRA model peer review was performed in 2013.

There have been numerous changes to the FPRA methodology since the last full scope peer review of the FPRA. The integration of NRC-accepted FPRA methods and studies described below that are relevant to this submittal could potentially impact the TSTF-505 results and/or the CDF and LERF. NRC has issued updated guidance for aspects of FPRA that supplant earlier guidance issued by NRC.

- NUREG-2180, "Determining the Effectiveness, Limitations, and Operator Response for Very Early Warning Fire Detection Systems in Nuclear Facilities," (DELORES-VEWFIRE)," (ML16343A058) regarding the updated approach to credit incipient fire detections systems.

- NUREG-2169, "Nuclear Power Plant Fire Ignition Frequency and Non-Suppression Probability Estimation Using the Updated Fire Events Database," (ML15016A069) regarding changes in fire ignition frequencies and non-suppression probabilities.

- NUREG/CR-7150, "Joint Assessment of Cable Damage and Quantification of Effects from Fire (JACQUE-FIRE)," Volume 2, "Expert Elicitation Exercise for Nuclear Power Plant Fire-Induced Electrical Circuit Failure," (ML14141A129) regarding possible increases in spurious operation probabilities.

- NUREG-2230, “Methodology for Modeling Fire Growth and Suppression Response for Electrical Cabinets Fires in Nuclear Power Plants,” (ML20157A148) regarding electrical cabinet fires.

- NUREG-2178, “Refining and Characterizing Heat Release Rates from Electrical Enclosures During Fire (RACHELLE-FIRE), (ML20168A655) regarding heat release rates (Volume 2).

Section 2.5.5 of RG 1.174, “An Approach for Using Probabilistic Risk Assessment in Risk-Informed Decisions on Plant-Specific Changes to the Licensing Basis,” provides guidance that indicates additional analysis is necessary to ensure that contributions from the above influences would not change the conclusions of the TSTF-505 LAR.

a) Provide a detailed justification for why the integration of the above NRC accepted FPRA methods and studies would not significantly impact the RICT calculation. As part of this justification, identify potential FPRA methodologies used in the FPRA that are no longer consistent with NRC guidance. Provide technical justification for methods in Quad Cities, Units 1 and 2, FPRA not accepted by the staff and evaluate the significance of their use on the RICT estimates.

OR

b) Alternatively, if the above guidance has been implemented in Quad Cities, Units 1 and 2, FPRA, provide the following:

    i. Indicate whether the changes to the FPRA are PRA maintenance or a PRA upgrade as defined in ASME/ANS RA-Sa-2009, Section 1-5.4, as qualified by RG 1.200, Revision 2, “Acceptability of Probabilistic Risk Assessment Results for Risk-Informed Activities,” (ML090410014) along with justification for the determination.

    ii. Discuss the focused scope (or full scope) peer review(s) that was performed to evaluate the changes that were determined in Part b.i. above to constitute a PRA upgrade and provide the date for when the peer review(s) was performed and for when the peer review report(s) that evaluated the incorporation of the method(s) was approved.

APLB RAI 11

RG 1.200, Revision 2 provides guidance for addressing PRA acceptability. RG 1.200, Revision 2, describes a peer review process using the ASME/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," as one acceptable approach for determining the technical acceptability of the PRA. The primary results of peer review are the 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, titled “NEI 05-04/07-12/12-06 Appendix X: Close-out of Facts and Observations (F&Os)” (ML17086A431), which was accepted by the NRC in a letter dated May 3, 2017 (ML17079A427).

Section 4 of Enclosure 2 to the TSTF-505 LAR states than one FPRA F&O, F&O 9-1, remains open and provides a succinct disposition. However, the TSTF-505 LAR does not provide the full description of the finding from the focused-scope peer review (FSPR), the recommendations from the FSPR team to address the finding, and a current disposition of this open F&O. Provide the FSPR full description, comments, recommendations, and licensee disposition related to this application for F&O 9-1.

APLC RAI 01

Section 2.3.1, Item 7 of NEI 06-09, states that the “impact of other external events risk shall be addressed in the RMTS program” and explains that one method to do this is by “performing a reasonable bounding analysis and applying it along with the internal events risk contribution in calculating the configuration risk and the associated RICT.” The NRC staff’s SE for NEI 06-09, states that “Where PRA models are not available, conservative, or bounding analyses may be performed to quantify the risk impact and support the calculation of the RICT.”

The revised flood analysis refers to a portable Darley pump to be utilized to provide makeup flow during certain external flooding scenarios. QCOA 0010-16, “Flood Emergency Procedure,” Rev. 29, states that to provide a suction source for the Darley pump, while waters are between 595 ft and 599 ft elevation, a hose should be routed over the 4 ft barrier installed in the Reactor Building (RB) ½ trackway personal access. Access to flood water suction provides Operations with an additional method of cooling makeup before flood waters exceed 599 ft.

However, this sequence does not appear to be discussed in the estimate of flood risk in the TSTF-505 LAR. The LAR provides an estimate of 0.3 conditional core damage probability (CCDP) for operators to install flood barriers.

a) Describe how the Darley pump mitigates risk during the external flood scenarios.

b) Provide the risk contribution (importance) of this sequence.

c) Describe the training operators receive regarding this scenario, including the frequency of training.

d) Describe how the timing for taking the action will be validated during training and how deviations between the actual and modeled required time will be addressed.

e) Describe the credit given for this scenario in the licensee’s RICT application.

f) Clarify if the licensee intends for this to be an implementation item for the applications. Provide, if needed, a description of the implementation item.

APLC RAI 02

The TSTF-505 LAR Attachment 4, Section 5, states that completion of EC 636914 “Update to LIP Barriers to Assist the Station External Flood Response,” which is being developed to modify barriers to protect the plant up to the 599 ft elevation, is tracked as a RICT Program

Implementation Item. It appears that EC 636912, “Update to the Station External Flood Response to Support Risk Reduction,” as well as changes to QCOA 0010-16 are not complete. The LAR states that “the addition of the LIP barrier installation to QC 0010-16 with available time will greatly reduce the risk to the station from external river floods.”

Clarify why the completion of these documents are not proposed as implementation items. Alternatively, provide implementation items that ensure completion prior to implementing either application.

APLC RAI 03

The TSTF-505 LAR states that installation of the Local Intense Precipitation (LIP) barriers in the to-be-revised QCOA-0010-16 requires 8 equipment operators working concurrently. QCOA 0010-22 states that “Fastlogs” weigh 113lbs and installation of each Fastlog requires a minimum of 2 people. The new flooding analysis states that operators will need additional training to support the flood strategy response where barriers are installed to prevent excessive water intrusion for floods up to 599 ft.

a) Clarify if licensee procedures account for the number of required equipment operators on site to perform this installation.

b) Provide the frequency of training that the operators receive or will receive on the to-be-revised procedure regarding this installation.

c) Describe how the timing for taking the action will be validated during training and how deviations between the actual and modeled required time will be addressed.

d) It appears that the Human Error Probability (HEP) and Conditional Core Damage Probability (CCDP) for this installation incorporates the failure of the barriers themselves.

   i. Describe the quality control measures for these barriers.

   ii. Discuss the barriers storage location.

   iii. Describe and justify the barriers failure rates.

APLC RAI 04

The licensee presentation during audit discussions on external flooding listed seven human failure events (HFEs), six with a human error probabilities (HEP) value of 5E-02 and one with a 3E-01 value. The 5E-02 value was stated as a conservative value assigned to HEP-1A (Tsw = 45 minutes), HEP-1B (Tsw = 45 minutes) , HEP-3 (Tsw = 5 minutes), HEP-8 (Tsw = 25 minutes), HEP-9 (Tsw = 40 minutes), and HEP-11 (Tsw = 45 minutes). Tsw represents the amount of time from the initiating event (T = 0) to complete the action. It is unclear to the NRC staff how HFEs with significantly different Tsw values and represent three types of installations (swing gates, panels, and fastlogs) would have the same HEP value. It is unclear what HFE XF-LIP-HEP represent that has a Tsw value of 45 minutes and a HEP of 3E-01.

Section 5.3.3 of NUREG 1792 provides good HRA practices for post-initiator HFEs. Good Practice #2 states no HFE screening value should be lower than 0.1 or lower than the worstcase anticipated value (which appears to be 3E-01). Good Practice #4 states to revisit the use of post-initiator screening values versus detailed assessments for special applications. It is unclear to the NRC staff if the above HFEs are developed or screened and if the assigned HEP value is appropriate and if screened that these HFEs should be adequately assessed. The NRC staff notes that Capability Category I of the 2009 ASME/ANS PRA Standard SR HR-G1 the use of screening values for HECC-I states to use conservative estimate for the HEPs of the HFEs. The presentation represented these values as conservative. It is unclear to the NRC staff if the HRA development of these HFEs are conservative or bounding.

a) Clarify what operator action HFE XF-LIP-HEP represents and why its probability is different than the other six HFEs.

b) Clarify if the seven HFEs listed associated with the barriers are screened or developed operator actions. For those HFEs that are screened, justify that they are not risk significant.

c) For the HFEs that have a screening value, that constitute different actions, and have different Tsw values, justify that the use of the same screening value is appropriate.

d) Provide justification that the HEP value of 5E-02 is consistent with the guidance of NUREG-1792.

e) Provide justification that these HEP values are conservative for this application consistent with the guidance in NEI-06-09-A.

Good Practice #6 of NUREG-1792 for post-initiator state to account for dependencies among post-initiator HFEs. SR HR-G7 requires, for multiple actions in the same accident sequence, to assess the degree of dependence and to calculate a joint HEP. It is unclear to the NRC staff if a dependency analysis (DA) was performed for the external flood sequence. Based on the material available to the staff during its regulatory audit, it appears that the following actions would also be performed during this sequence:

- Remove decay heat
- Install RHR 6" fire hose crossties to fire water supply
- Fill both torus
- Remove shiel plugs, drywell heads, and reactor vessel head
- Set up portable pump (Darley) for decay heat removal
- Fill Radwaste tanks with fire system water
- Portable makeup demineralizers to the CST
- Fill reactor cavities and separator-dryer pools
- Remove gates between storage pools
- Rack out all main breakers for equipment below 608 feet (lose normal decay heat removal systems)
- Open plant doors

The NRC staff understands that the licensee's analysis did not credit the portable pump. However it is unclear if the other actions listed above would be performed in addition to the barrier installations. and if all the relevant operator actions are included in the licensee's analysis. If the licensee's analysis relies on the arrival of offsite personnel, the staff notes that access to the plant (flooded roads and bridges) should be accounted for.

f) Clarify if a DA was included in the external combined effects flood. If not, provide justification for its exclusion.

g) Clarify what other operator actions would be required during the installation of the barriers. If yes, include in this discussion their inclusion in the DA.

h) Clarify if offsite personnel are required for this response. If yes, include in this discussion how plant access was considered.

i) Based on the responses to Parts (d), (e), and (f) provide justification that sum of these issues does not significantly impact the application.

APLC RAI 05

Material available to the staff during its regulatory audit (specifically, document EC 636914) states that the RB crane will be loaded onto the EDG if a LOOP occurs during external flooding.

a) Describe the function of the RB crane in this scenario.

b) Is loading the RB crane onto the EDG proceduralized as part of the EDGs loading following LOOP? If not, how will the loading be ensured if such loading is required.

APLC RAI 06

The NRC staff reviewed the licensee's integrated assessment (IA; ML18180A033) for the revised flood hazard as part of the agency's post-Fukushima actions. The staff's review is documented at ML19168A196.

a) Identify instances where the revised flooding analysis and actions provided for this application change the information provided to the staff as part of the licensee's IA.

b) If changes to the licensee's IA are identified, for each change, justify why the staff's conclusions on the IA continue to remain valid.

APLC RAI 07

NEI 00-04[3] Figure 5-6 provides guidance to be used to determine SSC safety significance. The same document states, in part, that if 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 LSS category.

Section 3.2.4 of the 50.69 LAR states that "All external hazards, except for seismic, were screened for applicability to QCNPS [Quad Cities Nuclear Power Station] using a plant-specific evaluation in accordance with Generic Letter 88-20 (Reference [27]) and updated to use the criteria in ASME PRA Standard RA-Sa-2009."

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[3] NEI 00-04, Revision 0, "10 CFR 50.69 SSC Categorization Guideline", July 2005 (ADAMS Accession No. ML052910035).

Likewise, Attachment 4 of the 50.69 LAR lists all external hazards as screened except for seismic hazard with an alternate approach. The guidance in NEI 00-04, Figure 5-6 regarding SSCs that play a role in screening a hazard is not discussed in Section 3.2.4 nor in Attachment 4 of the LAR. Therefore, it appears to the NRC staff based on this lack of information that at the time an SSC is categorized it will not be evaluated using the guidance in NEI 00-04, Figure 5-6 to confirm that the SSC is not credited in screening an external hazard because that evaluation has already been made. NRC staff notes that plant changes, plant or industry operational experience, updates to hazard frequency information, and identified errors or limitations in the PRA models could potentially impact the conclusion that an SSC is not needed to screen an external hazard.

a) Clarify whether an SSC will be evaluated during categorization of the SSC using the guidance in NEI 00-04, Figure 5-6 to confirm that the SSC is not credited in screening an external hazard.

b) If an SSC will not be evaluated using the guidance in NEI 00-04, Figure 5-6 to confirm that the SSC is not credited in screening an external hazard at the time of categorization because that evaluation has already been made, then explain how plant changes, plant or industry operational experience, updated information in hazard frequencies, and identified errors or limitations that could change that decision are addressed.

Table in Attachment 4 of the 50.69 LAR under the external flooding evaluation, states "Table A4-1 includes a list of LIP barriers credited for screening this hazard." Table A4-1 lists 14 barriers credited for screening the external flood hazard (six of which require manual action). The TSTF-505 LAR the states that "the addition of the LIP barrier installation to QC 0010-16 with available time will greatly reduce the risk to the station from external river floods."

c) Describe how the SSCs listed in Table A4-1 are to be categorized using the guidance in NEI 00-04, Figure 5-6.

<u>APLC RAI 08</u>

NEI 00-04, Rev. 0, Section 5, "Component Safety Significance Assessment" states, "If the plant does not have an external hazards PRA, then it is likely to have an external hazards screening evaluation that was performed to support the requirements of the IPEEE." NEI 00-04, Rev. 0, also states in Section 3.3.2, "Other Risk Information (including other PRAs and screening methods)," that the characterization of the adequacy of risk information should include "a basis for why the other risk information adequately reflects the as-built, as-operated plant."

The TSTF-505 and 10 CFR 50.69 LARs, Table E4-16 and Attachment 4 respectively, state under the Industrial or Military Facility section that "none of the operations at Cordova Industrial Park pose any threat to QCNPS from explosion, explosive shock, resulting missiles, or toxic fumes release" yet there is no mention of the effect of the CF Industries Chemical complex on the plant. Please explain the impact of the CF Industries Chemical complex on the plant and either justify that the impact can be screened for this application or describe how the impact is included in the RICT program.

APLC RAI 09

The TSTF-505 and 10 CFR 50.69 LARs, Table E4-16 and Attachment 4 respectively, states that criterion “C4” (event is included in the definition of another event) and criterion “C5” (event develops slowly, allowing adequate time to eliminate or mitigate the threat) was used to screen the snow hazard. The LARs focus on potential flooding impacts, but not on the design basis roof live load or the maximum recorded snowfall for the site. It is unclear to the NR staff whether the risk of this hazard is adequately considered for this application.

Justify the screening of risk associated with snowfall from the application (e.g., by comparing historical maximum snowfall against the design basis, showing that the occurrence frequency of snowfall events that could challenge the plant is low).

EEEB RAI-01

According to the Quad Cities, Units 1 and 2, Updated Final Safety Analysis Report, chapter 8, page 8.3-9, the plant has three emergency diesel generators (EDGs), one dedicated to each unit, and a common which is automatically connected to the unit in which loss of offsite power (LOOP) and loss-of-coolant accident (LOCA) occurs. Considering this arrangement, explain whether the plant (both units) can be safely shutdown in the following scenarios: LOOP for both units, a LOCA on a unit, and a single failure of EDG on another unit. If not, please explain the single-failure criteria as applied to the two units. This clarification will help understand/verify the Design Success Criteria considered in table E-1 of the LAR.

EEEB RAI-02

According to TSTF-505 LAR, table E1-1, corresponding to TS Condition 3.8.1.B “One required DG inoperable,” the design success criteria require “Two of out of three DGs”, whereas the PRA success criteria require “One out of three DGs.” Explain the reason for this difference.

EEEB RAI-03

Explain the single-failure criteria considered for the safety-related 250 VDC and 125 VDC systems as applied to the two units.

EEEB RAI-04

In the LAR, corresponding to TS Conditions 3.8.7.A, “One or more AC electrical power distribution subsystems inoperable” the proposed RICT program has a NOTE stating: “Only applicable when a loss of function has not occurred.” Explain why a similar NOTE is not proposed for the TS Condition 3.8.7.B, “One or more DC [direct current] electrical power distribution subsystems inoperable” and TS Condition 3.8.7.C, “One or more required opposite unit AC or DC electrical power distribution subsystems inoperable.”

EEEB RAI-05

According to the LAR, table E1-1, corresponding to the TS Condition 3.8.1.C “Two required offsite circuits inoperable,” the design success criteria require “Two required offsite circuits.” Explain how the design success criteria will be met with “Two required offsite circuits inoperable.”

EEEB RAI-06

According to the LAR, table E1-1, the TS Condition 3.8.1.D states “One required offsite circuit inoperable OR one required DG inoperable.” Please confirm that the TS Condition 3.8.1.D in table E1-1 was meant to state “One required offsite circuit inoperable AND one required DG inoperable.” Also, according to the table E1-1, for the design success require “Two required sources.” Explain the two required sources.

EICB RAI 01

In Section 3.1.2.3 “Evaluation of Instrumentation and Control Systems” of the TSTF-505 Revision 2 Model Safety Evaluation, the NRC clarifies that the basis of the staff’s safety evaluation is to consider “a number of potential plant conditions allowed by the new TSs” and to consider “what redundant or diverse means were available to assist the licensee in responding to various plant conditions.” The TSTF-505 Revision 2 position recommends that “at least one redundant or diverse means (e.g., other automatic features or manual action) to accomplish the safety functions (e.g., reactor trip, safety injection, or containment isolation) remain available during the use of the RICT.” This approach is consistent with maintaining a sufficient level of defense-in-depth in accordance with RG 1.174, Revision 2, “An Approach for Using Probabilistic Risk Assessment in Risk Informed Decisions on Plant Specific Changes to the Licensing Basis,” (ML100910006), and the guidance in Revision 1 of RG 1.177, “An Approach for Plant Specific, Risk Informed Decisionmaking: Technical Specifications,” (ML100910008), which further describe the regulatory position with respect to defense-in-depth (including diversity).

Enclosure 1 of the TSTF-505 LAR lists the functions of the Instrumentation and Control Systems and their design logics; however, this list does not provide NRC staff adequate information to verify that at least one redundant or diverse means will remain available to accomplish the intended I&C safety functions during the proposed risk informed completion time.

Describe other means that exist to initiate the safety function for each plant accident condition that each affected I&C function is currently designed to address. The evaluation of “diverse means,” should identify the conditions that the functional unit responds to, and for each condition, other means (e.g., diversity, redundancy, or operator actions) that can be used. Alternatively, provide additional information to demonstrate that defense-in-depth is maintained during the extended completion times for each function. This information is needed to demonstrate compliance with 10 CFR 50.36(c), and consistency with the implementing guidance in RG 1.174 and the TSTF-505, Revision 2.

STSB RAI-01

In Attachment 2 of the TSTF-505 LAR, the proposed change to add RICTs for Quad Cities TSs Required Action 3.3.5.1.F.2 (place channel in trip) is, in part:

> 96 hours from discovery of inoperable channel concurrent with HPCI [high-pressure coolant injection] or reactor core isolation cooling (RCIC) inoperable
> …
> OR
>
> In accordance with the Risk-Informed Completion Time Program
> …

The NRC staff recognizes that the licensee's proposed change is consistent with the NUREG-1433 "Standard Technical Specifications General Electric BWR/4 Plants," Revision 4 (ML21272A358) TS markups in TSTF-505, Revision 2. However, it has been brought to staff's attention that some of the TSTF-505 markups contain errors, introducing potential for licensee actions to be less conservative than the original intent of the requirements. To modify completion times that include the phrase "from discovery," the RICT shall start at discovery instead of the time the TS action statement is entered, or the normal "time zero." The requirement is not clear when the RICT statement is separated from the "from discovery" statement.

To provide clarity, discuss how the proposed change ensures that the time of entry for the condition as at from discovery is clear or revise the placement of the proposed completion time for TS Required Action 3.3.5.1.F.2 between "96 hours" and "from discovery." Also, provide a similar discussion or change for proposed revision to Required Action 3.3.5.1.G.2, which is formatted similarly.

<u>STSB RAI-02</u>

In TSTF-505, Revision 2, TS Example 1.3-8 appears to contain a typographical error. Assess whether the reference in the first paragraph to "Condition C" should be revised to "Condition B."

<u>STSB RAI-03</u>

Discuss why the following Required Actions have a proposed note that excludes loss of function (LOF) conditions for the RICT program when the associated LCO has a separate TS condition that addresses a LOF condition. (This question is seeking to understand, not necessarily to revise, the proposed TS.)

a. RA 3.3.2.2.A.1 (TS 3.3.2.2 Condition B addresses LOF)
b. RA 3.3.4.1.A.1 and A.2 (TS 3.3.4.1 Conditions B and C address LOF)
c. RA 3.3.6.1.A.1 (TS 3.3.6.1 Condition B addresses LOF)
d. RA 3.8.7.A.1 (TS 3.8.7 Condition E addresses LOF)

<u>STSB RAI-04</u>

For TS Required Action 3.8.4.B.2, the current completion time is "[p]rior to exceeding 7 cumulative days per operating cycle of battery inoperability, on a per battery basis, as a result of maintenance or testing." The frontstop "7 cumulative days…" is not fixed like all described in TSTF-505, Revision 2; it is variable depending on how many days have been used during the operating cycle for an inoperable battery. Applying a RICT to this required action is also inconsistent with the guidance in NEI 06-09-A that does not mention variable frontstop completion times.

Provide justification for the following questions or remove this proposed change:

a) How would the variable frontstop be evaluated to begin calculating a RICT?
    i. Provide an example where more than 1 day but less than 7 days has been applied to an inoperable battery in the operating cycle.
    ii. Provide an example of what would happen when TS 3.8.4 Condition B is applicable and the 7-day threshold has already been exceeded (e.g., RICT was used one time and exited during the same operating cycle).

b) Are there ever emergent maintenance and testing scenarios that would apply to TS 3.8.4 Condition B? If so, would a RICT be applied after the 7-day threshold were exceeded and what frontstop value would be selected?

<u>STSB RAI-05</u>

The proposed administrative controls for the RICT program in TS 5.5.15 paragraph “e” of Attachment 2 of the TSTF-505 LAR was based on the TS markups of TSTF-505, Revision 2, for Quad Cities, Units 1 and 2. The NRC staff recognizes that the model safety evaluation (SE) for TSTF-505, Revision 2, contains improved phrasing for the administrative controls for the RICT program in TS 5.5.15 paragraph “e,” namely the phrasing “approved for use with this program” instead of “used to support this license amendment.” In lieu of the original phrasing in TS 5.5.15 paragraph “e”, discuss whether the phrases “methods used to support Amendment # xxx” or, as discussed in the TSTF-505 model SE, “methods approved for use with this program” would provide more clarity for this paragraph.