From: Green, Kimberly
Sent: Friday, August 19, 2022 11:55 AM
To: Eckermann, Jeremy Beau
Cc: Victor, William Ross
Subject: Browns Ferry TSTF-505 Audit Questions (EPID L-2022-LLA-0049)
Attachments: BFN TSTF-505 Audit Questions.docx

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Beau,

The June 14, 2022, Audit Plan (ML22147A137) states that "[t]hroughout the audit period, the NRC staff will submit audit questions and audit-related requests to TVA so that TVA can better prepare for audit discussions with NRC staff." Attached is the first set of NRC staff audit questions.

We look forward to discussing these questions and TVA's responses during the virtual audit meeting that is currently scheduled to take place during the week of September 12, 2022. Please contact me at any time prior if a clarification discussion is needed.

Please post the responses for the questions to the Certrec Portal as the response is completed.

Let me know if you have any questions.

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AUDIT QUESTIONS
LICENSE AMENDMENT REQUEST
TO REVISE TECHNICAL SPECIFICATIONS
TO ADOPT TSTF-505, REVISION 2

TENNESSEE VALLEY AUTHORITY

BROWNS FERRY NUCELAR PLANT UNITS 1, 2, AND 3

DOCKET NOS. 50-259, 50-260, AND 50-296

Tennessee Valley Authority (TVA) submitted a license amendment request (LAR) for Browns Ferry Nuclear Plant (BFN), Units 1, 2, and 3.[1] The amendments would revise certain technical specifications to permit the use of a risk-informed completion time (RICT) for actions to be taken when a limiting condition for operation is not met. The proposed changes to implement RICT are based on a Technical Specifications Task Force (TSTF) Traveler, TSTF-505, Revision 2, "Provide Risk-Informed Extended Completion Times."[2] The LAR also proposed to implement TSTF-439, "Eliminate Second Completion Times Limiting Time From Discovery of Failure To Meet an LCO."[3]

The NRC staff will audit TVA's records that support these requests. Please address the following questions during the audit.

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1 Polickoski, J.T., TVA, letter to the NRC, "License Amendment Request to Revise Technical Specifications to Adopt Risk-Informed Completion Times" dated March 31, 2022 (Agencywide Documents Access and Management System (ADAMS) Accession No. ML22090A287).

2 TSTF-505, Revision 2, "Provide Risk-Informed Extended Completion Times – RITSTF Initiative 4b," dated July 2, 2018 (ML18183A493)

3 TSTF-439, Revision 2, "Eliminate Second Completion Times Limiting Time From Discovery of Failure To Meet an LCO," dated June 20, 2005, which has been incorporated in standard technical specifications (NUREG-1433, "Standard Technical Specifications—General Electric Plants (BWR/4)," Revision 5, September 2021 (ML21272A357).

Audit Questions from Probabilistic Risk Assessment Licensing Branches A and C (APLA/APLC)

Question 1 (APLA/APLC) – Consideration of shared systems in calculation of a RICT

Regulatory guidance states that "the PRA results used to support an application must be derived from a baseline PRA model that represents the as-built, as-operated plant to the extent needed to support the application."[1]

The license amendment request does not appear to address the existence of crossties between units.[2] However, the NRC staff has reviewed system documents in the portal that have shared systems. The staff notes that for some of these systems it appears the sharing of a system is not consistent among units. It appears that some operational aspects, such as alternate alignments, were excluded from the PRA models.

Clarify which systems are shared, how they are shared, whether they are capable of supporting the other units in an accident. Explain how the shared systems are credited for each unit in the

PRA models. The NRC staff notes that for multiunit events (e.g., loss of offsite power and seismic events), credit for a system may be limited to one unit.

a) Identify systems that can be crosstied to another unit. Discuss any differences among the units sharing these systems.

b) Explain how shared systems credited in the real-time risk model that support the RICT calculations are modeled for each unit in a multiunit event. Include in this discussion what aspects of these systems were excluded from the PRA model(s) and why these exclusions do not impact the application.

c) If the impact of events that can create a concurrent demand for a system shared by multiple units and credited in the real-time risk model is not addressed, explain why this modeling exclusion does not have a significant effect on the RICT calculations.

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1 NRC Regulatory Guide 1.200, "An Approach for Determining the Technical Adequacy of Probabilistic Risk Assessment Results for Risk-Informed Activities," Revision 2, dated March 2009 (ML090410014)

2 Polickoski, J.T., TVA, letter to the NRC, "License Amendment Request to Revise Technical Specifications to Adopt Risk-Informed Completion Times" dated March 31, 2022 (ML22090A287).

Question 2 (APLA/APLC) – PRA Model Uncertainty Analysis Process

The NRC approved the industry guidelines for risk-managed technical specifications.[1] The NRC staff's safety evaluation of this guidance for specifies that a LAR to allow the use of a risk-informed completion time (RICT) should include identification of key assumptions and sources of uncertainty.[2] Each one should be assessed and dispositioned as to its effect on the application. The NRC has published guidance on the process of identifying, characterizing, and qualitatively screening model uncertainties.[3]

Section 6 of Enclosure 9 to the LAR presents five screening criteria used to determine if a source of uncertainty is key to this application.[4] The NRC staff notes that none of the criteria specify the use of sensitivity studies to determine the actual impact of the uncertainty on PRA results, specifically on RICT calculations. Also, the sources of uncertainty listed in Tables E9-1 through E9-3 address only items that are not included in the model. It is unclear to the staff how any potential source of uncertainty was determined not to impact this application.

a) Discuss the criteria used to determine when a sensitivity study was needed to address the identified source of uncertainty.

b) Discuss any sensitivity studies performed that determined the impact of sources of uncertainty on RICT calculations. Address why each identified source of uncertainty is not key to this application.

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1 NEI 06-09-A, Risk-Managed Technical Specifications Guidelines," November 2006 (ML12286A322).

2 Letter from J.M. Golder (NRC) to B. Bradley (NEI), "Final safety evaluation for Nuclear Energy Institute Topical Report NEI 06-09," dated May 17, 2007 (ML071200238).

3 NUREG-1855, Revision 1, "Guidance on the Treatment of Uncertainties Associated with PRAs in Risk-Informed Decision Making, Main Report," dated March 2017 (ML17062A466)

4 Polickoski, J.T., TVA, letter to the NRC, "License Amendment Request to Revise Technical Specifications to Adopt Risk-Informed Completion Times" dated March 31, 2022 (ML22090A287).

Question 3 (APLA/APLC) – PRA Model Uncertainty Analysis for Updated PRA Models

Section 2 of Enclosure 5 to the LAR identifies the model of record (MOR) for each PRA used in the application.[1] For the internal events PRA it is MOR 10, for the fire PRA it is MOR 6, and for the seismic PRA it is MOR 1. The NRC staff reviewed the portal documents provided and it appears that some uncertainty analyses were conducted using earlier models. The NRC staff notes that an MOR update can significantly affect the risk results. It may raise the importance of a previously analyzed source of uncertainty that was not significant to the point where it becomes a key source of uncertainty. A model update can also introduce a new source of uncertainty.

a) Clarify how the uncertainty analyses that the staff has audited reflect the MORs used to support this application.

b) For any PRA hazard model that has a gap between the last uncertainty review and the MOR used for this application, justify the conclusion that no key source of uncertainty relevant to this application has been introduced.

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[1] Polickoski, J.T., TVA, letter to the NRC, "License Amendment Request to Revise Technical Specifications to Adopt Risk-Informed Completion Times" dated March 31, 2022 (ML22090A287).

Question 4 (APLA) – In-Scope LCOs and Corresponding PRA Modeling

The LAR should provide a comparison of the technical specification functions to the PRA modeled functions to show that the PRA modeling is consistent with the licensing basis assumptions.[1] If there is a difference, it should be explained. Table E1-1 in Section 1 of Enclosure 1 to the LAR identifies each limiting condition for operation (LCO) in the technical specifications proposed for inclusion in the RICT program.[2] 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 3.3.5.1.B, Table E1-1 states that the suppression pool valves will be used as surrogates for the Reactor Vessel Level Low – Level 0 instrument signal (Function 2.e). It is unclear to the NRC staff which systems are affected by the Function 2.e signal.

   i. Describe the systems that are affected by the Function 2.e signal. In this description, address each configuration of the systems and how they are affected.

   ii. Explain how the suppression pool surrogate bounds these effects.

b) Regarding TS 3.3.6.1.A, Table E1-1 states for Function 3 (HPCI isolations), Function 4 (RCIC isolations), Function 5 (RWCU isolations), and Function 6 (SDC isolations) that one of the modeled pathways will be used as a surrogate. It is unclear to the NRC staff which pathways will be used for each of these functions.

   i. Clarify which modeled pathways will be used as a surrogate for each of the system isolation functions.

   ii. Explain how the surrogate bounds each of the isolation functions.

c) Regarding TS 3.6.1.2.C, Table E1-1 states that one of the modeled pathways will be used as a surrogate when one of the primary containment airlocks is inoperable. Please indicate the impact of this surrogate on large early release calculations compared to the airlock.

   i. Briefly describe the effect of the failure of early containment isolation (i.e., plant response to the failure of the modeled pathway).

   ii. Explain how this bounds the effect of an inoperable containment airlock door.

d) Regarding TS 3.6.1.3.A, Table E1-1 states that, for the valves not modeled, a pathway that is modeled will be used as a surrogate. It is unclear to the NRC staff which pathways will be used for each affected function.

   i. Clarify which modeled pathways will be used as a surrogate for each of the system isolation functions affected.

   ii. Provide justification that the surrogate bounds each of the isolation functions.

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[1] Letter from J.M. Golder (NRC) to B. Bradley (NEI), "Final Safety Evaluation for Nuclear Energy Institute Topical Report NEI 06-09," dated May 17, 2007 (ML071200238).

[2] Polickoski, J.T., TVA, letter to the NRC, "License Amendment Request to Revise Technical Specifications to Adopt Risk-Informed Completion Times" dated March 31, 2022 (ML22090A287)

Question 5 (APLA/APLC) – Update of Fire and Seismic PRAs with the resolution of Internal Event F&Os

Regulatory guidance on PRA technical adequacy calls for peer review of the PRA model and its results.[1] The primary results of peer review are the facts and observations (F&Os) recorded by the peer review team and the subsequent resolution of these F&Os using the closure process documented in industry guidance that the NRC has accepted for use.[2]

The NRC staff notes that implementation of risk-informed categorization of SSCs under 10 CFR 50.69 has been authorized by license amendments issued to BFN.[3] A prerequisite to implementation of these amendments is that "the resolutions to the internal events findings with the potential to impact the FPRA and SPRA modeling will be incorporated into the FPRA and SPRA." Enclosure 2 to the LAR does not discuss the resolution of internal events F&Os in either the fire or seismic PRA models.[4] It is not clear whether the PRA models to be used in RICT calculations have incorporated all relevant F&O resolutions.

a) Confirm that all internal events PRA modeling updates performed to resolve F&Os that could impact fire or seismic risk were incorporated into the fire and seismic PRA models.

b) Alternatively, propose a mechanism that ensures that all internal events modeling updates performed to resolve F&Os that could affect fire or seismic risk are incorporated into the affected PRA models prior to implementation of the RICT program. Alternatively, justify that the internal events modeling updates performed to resolve F&Os have no consequential effect on the RICT calculations.

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[1] NRC Regulatory Guide 1.200, "An Approach for Determining the Technical Adequacy of Probabilistic Risk Assessment Results for Risk-Informed Activities," Revision 2, dated March 2009 (ML090410014)

[2] NEI 05-04/07-12/12-06 Appendix X, "Close-out of Facts and Observations (F&Os)," dated February 21, 2017(ML17086A431).

[3] Three license amendments: No. 317 for Renewed License DPR-33, No. 340 for Renewed License DPR-52, and No. 300 for Renewed License DPR-68 dated July 27, 2021 (ML21173A177)]

4 Polickoski, J.T., TVA, letter to the NRC, "License Amendment Request to Revise Technical Specifications to Adopt Risk-Informed Completion Times" dated March 31, 2022 (ML22090A287)

Question 6 (APLA) –Risk Calculation Based on the Model of Record

Regulatory guidance provides risk acceptance criteria for total core damage frequency (CDF ≤ 1E-04/year) and large early release frequency (LERF ≤ 1E-05/year).[1] The NRC has also published guidance stating that for a capability category II risk evaluation, the mean values of the risk metrics (total risk and change in risk) are evaluated against these guidelines.[2] More specifically, these metrics are the means of the probability distributions that result from the propagation of input parameter uncertainties and model uncertainties explicitly reflected in the PRA models.

In general, the point estimates of CDF and LERF obtained by quantification of the cutset probabilities using mean values for each basic event probability do not represent true mean values of the CDF and LERF. Under certain circumstances, a formal propagation of uncertainty may not be required if it can be demonstrated that the state-of-knowledge correlation (SOKC) is unimportant (i.e., the risk results are well below the acceptance guidelines).

Table E5-1 of Enclosure 5, Section 2 of Attachment 1 to the LAR presents point estimates of CDF and LERF based on the optimized one-top multihazard model.[3] For each of the three units, values for total risk are listed along with risk contributions from internal events (which includes internal flooding), internal fire, and seismic events. These reported values did not match the point estimates developed from the model of record.

When quantification explicitly addresses the SOKC, point estimates are likely to be lower than the mean risk values. The total LERF is reported as 9.08E-06/year, 8.63E-06/year, and 8.53E-06/year for BFN Units 1, 2, and 3, respectively. When both model updates to resolve F&Os and the SOKC are considered, the total risk could approach the criteria from RG 1.174.

a) Demonstrate that the total risk for Units 1, 2, and 3 meet RG 1.174 risk acceptance guidelines for CDF and LERF. Mean values based on the model of record should be appropriately combined to include risk from internal events, seismic hazards, and fire as well as accounting for the SOKC. In this demonstration, account for changes in risk due to PRA model updates needed in response to NRC staff information requests or previous commitments and include identification of the parameters that are assumed to be correlated in the parametric uncertainty analysis of fire and seismic events.)

b) Alternatively, propose a mechanism that will ensure that total risk will satisfy RG 1.174 risk acceptance guidelines for CDF and LERF prior to implementation of the RICT program.

1 Regulatory Guide 1.174, Revision 3, "An Approach for Using Probabilistic Risk Assessment in Risk-Informed Decisions on Plant-Specific Changes to the Licensing Basis," January 2018 (ML17317A256).

2 Section 6.4 of NUREG-1855, Revision 1, "Guidance on the Treatment of Uncertainties Associated with PRAs in Risk-Informed Decision Making" dated March 2017 (ML17062A466).

3 Polickoski, J.T., TVA, letter to the NRC, "License Amendment Request to Revise Technical Specifications to Adopt Risk-Informed Completion Times" dated March 31, 2022 (ML22090A287).

Question 7 (APLA) – Consistency of the Results Produced by the CRM Tool With the PRA Model Of Record

The level of detail in the PRA should be sufficient to model the impact of the proposed licensing basis change.[1] The characterization of the problem should include establishing a cause-effect relationship to identify portions of the PRA affected by the issue being evaluated. Full-scale applications of the PRA should reflect this cause-effect relationship when quantifying the effect of the proposed licensing-basis change on the PRA elements.

Industry guidance describes attributes of the tool used for configuration risk management (CRM).[2] The PRA models are normally transformed to create a real-time risk (RTR) model suitable for use with the CRM tool. Appropriate benchmarking of results from the CRM tool against the PRA model shall be performed to demonstrate consistency.

The staff reviewed how the one-top multihazard model (OTMHM) was integrated with the Phoenix model and how example evaluations were used to check the Phoenix model. However, it is not clear how results using the CRM tool were directly benchmarked against results from the PRA MOR. The NRC staff noted that benchmarking was performed between the individual PRA hazard models on one hand and the OTMHM on the other, with each hazard quantified separately. Total CDF and total LERF were quantified for each unit, using the OTMHM.

c) Clarify how the benchmarking activities performed can confirm consistency of the RTR model results with the results of each PRA model of record.

d) Describe how updates to the RTR model will maintain alignment with the PRA model of record.

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1 Regulatory Guide 1.174, Revision 3, "An Approach for Using Probabilistic Risk Assessment in Risk-Informed Decisions on Plant-Specific Changes to the Licensing Basis," January 2018 (ML17317A256), Regulatory Position 2.3.3.

2 Section 4.2 of NEI 06-09-A, Risk-Managed Technical Specifications Guidelines," November 2006 (ML12286A322).

Audit Questions from Probabilistic Risk Assessment Licensing Branch B (APLB)

Question 8 (APLB) – Fire PRA Uncertainty Associated with Methods

The NRC staff evaluates the acceptability of the PRA for each new risk-informed application. The technical acceptability of risk analyses that support a regulatory decision may vary with the relative weight given to the risk assessment element of the decision-making process.[1] The NRC staff notes that the calculated results of the PRA are used to determine a RICT, which limits how long SSCs controlled by technical specifications can remain inoperable. (This applies to individual SSCs as well as multiple, unrelated SSCs.) Accordingly, the PRA results are given a very high weight in a RICT application. The NRC staff requests additional information on issues that have been previously identified as potentially key assumptions for the FPRA.

The NRC staff notes that since the initial BFN NFPA 805 license amendment request,[2] the NRC has issued guidance (cited in the footnotes) that can affect the fire PRA modeling. These reports (most of them prepared jointly with EPRI) address several issues:

- fire ignition frequency and non-suppression probability [3]
- cabinets, motor, transformers, and control boards [4] [5]
- an updated approach to credit incipient fire detection systems [6]
- electrical cabinet fires [7]
- fire-induced electrical circuit failure [8]

a) Describe whether and how the cited fire PRA guidance has been incorporated into the BFN fire PRA. As applicable, summarize the changes and indicate whether a particular change was a PRA maintenance activity or a PRA upgrade as defined in the PRA Standard (specifically, Section 1-5.4)[9] as endorsed by the NRC.[10] Explain how this determination was reached. If any change constituted a PRA upgrade, describe the peer review of the change, report any open facts and observations (F&Os), and provide documentation of their disposition in accordance with industry guidance.[11]

b) If the fire PRA guidance identified above has not been incorporated into the BFN fire PRA, explain why application of the fire PRA guidance would have no adverse impact on calculations of risk and RICT.

c) Identify any fire PRA methodologies used in the BFN fire PRA that are no longer accepted by the NRC staff. If such a methodology was used, provide technical justification for its use in the fire PRA supporting RICT calculations and evaluate the significance of its use on reported risk estimates.

d) Alternatively, propose a mechanism that will ensure that the identified issues will be adequately addressed in the BFN fire PRA prior to implementation of the RICT program.

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1 Regulatory Guide 1.174, Revision 3, “An Approach for Using Probabilistic Risk Assessment in Risk-Informed Decisions on Plant-Specific Changes to the Licensing Basis,” January 2018 (ML17317A256).

2 Shea, J.W., Tennessee Valley Authority, letter to U.S. Nuclear Regulatory Commission, "Browns Ferry Nuclear Plant, Units 1, 2, and 3, License Amendment Request to Adopt NFPA 805 Performance-Based Standard for Fire Protection for Light Water Reactor Electric Generating Plants (2001 Edition) (Technical Specification Change TS-480)," dated March 27, 2013 (ML13092A393).

3 NUREG-2169, “Nuclear Power Plant Fire Ignition Frequency and Non-Suppression Probability Estimation Using the Updated Fire Events Database,” January 2015 (EPRI 3002002936, ML15016A069)

4 NUREG-2178, “Refining And Characterizing Heat Release Rates From Electrical Enclosures During Fire (RACHELLE-FIRE),” Volume 1, “Peak Heat Release Rates and Effect of Obstructed Plume, Final Report,” April 2016 (EPRI 3002005578, ML15114A477, ML15266A516).

5 NUREG-2178, “Refining And Characterizing Heat Release Rates From Electrical Enclosures During Fire (RACHELLE-FIRE),” Volume 2, “Fire Modeling Guidance for Electrical Cabinets, Electric motors, Indoor Dry Transformers, and the Main Control Board,” June 2020 (EPRI 3002016052).

6 NUREG-2180, “Determining the Effectiveness, Limitations, and Operator Response for Very Early Warning Fire Detection Systems in Nuclear Facilities (DELORES-VEWFIRE), Final Report,” December 2016 (ML16343A058).

7 NUREG-2230, “Methodology for Modeling Fire Growth and Suppression Response for Electrical Cabinet Fires in Nuclear Power Plants,” June 2020 (EPRI 3002016051, ML20157A148).

8 NUREG/CR-7150, “Joint Assessment of Cable Damage and Quantification of Effects from Fire (JACQUE-FIRE): Final Report,” Volume 2, “Expert Elicitation Exercise for Nuclear Power Plant Fire-Induced Electrical Circuit Failure,” May 2014 (BNL-NUREG-98204-2012, EPRI 3002001989, ML14150A294).

9 Joint Committee on Nuclear Risk Management of ASME and the American Nuclear Society, ASME/ANS RA-Sa–2009, “Addenda to ASME/ANS RA-S–2008, ‘Standard for Level 1/Large Early Release Frequency Probabilistic Risk Assessment for Nuclear Power Plant Applications,’” February 2009.

10 Regulatory Guide 1.200, Revision 3, "Acceptability of Probabilistic Risk Assessment Results for Risk-Informed Activities," December 2020 (ML20238B871)

11 NEI 05-04/07-12/12-06 Appendix X, “Close-out of Facts and Observations (F&Os),” dated February 21, 2017(ML17086A431).

Audit Questions from Containment and Plant Systems Branch (SCPB)

Question 9 (SCPB) – Residual Heat Removal Function

Section 2.3.2.3 of Attachment 1 to the LAR states that a RICT can be applied to LCO 3.7.1, Required Action D.1, because the remaining operable residual heat removal (RHR) service water (RHRSW) subsystems are adequate to perform the RHRSW heat removal function.[1] Technical Specification 3.7.1, Condition D is "two RHRSW subsystems inoperable." As described in the Browns Ferry TS Bases, the RHRSW system has four subsystems, with two pumps in each subsystem. With this configuration, a single active failure will ensure that there are at least two subsystems, with one pump operating in each subsystem, will perform the required cooling function.

The NRC staff notes that RHRSW is a shared system between the three Browns Ferry units. Technical Specification LCO 3.7.1 requires four RHRSW subsystems be operable. Condition D of LCO 3.7.1 allows two RHRSW subsystems to be inoperable for up to 7 days. The remaining two subsystems perform the RHR heat removal function for three Browns Ferry units. TVA indicated in TS Bases that under the configuration of Condition D (two subsystems inoperable), the RHRSW system will perform the required cooling function for three units without a loss of function (LOF). An LOF review is part of a defense-in-depth review in TSTF-505 for justifying the applicability of RICT.

The regulation at 10 CFR 50.36(b) states that technical specifications will be derived from the analyses and evaluation in the safety analysis report and amendments thereto (i.e., the updated final safety analysis report (UFSAR)). The NRC staff was unable to find such an analysis in the UFSAR to support LCO 3.7.1, Condition D, that two RHRSW subsystems will perform the intended safety function without an LOF.

Identify the applicable UFSAR sections or other reference documents or provide the specific safety analysis that demonstrate that two RHRSW subsystems are adequate to perform the intended safety function of RHR heat removal for the three units without an LOF.

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[1] Polickoski, J.T., TVA, letter to the NRC, "License Amendment Request to Revise Technical Specifications to Adopt Risk-Informed Completion Times" dated March 31, 2022 (ML22090A287)

Audit Questions from Technical Specifications Branch (STSB)

Question 10 (STSB) – Example of Emergent Failure

Tables E1-2, E1-3, and E1-4 of Enclosure 1 to the LAR contain example RICT calculations for Browns Ferry, Units 1, 2, and 3, respectively.[1] For TS Conditions 3.3.1.1.A, 3.3.1.1.B, and 3.3.6.1.A, the RICT estimates state "No Voluntary Entry," with a footnote that states:

> (2) Per NEI 06-09-A, for cases where the total CDF or LERF is greater than 1E-03/yr or 1E-04/yr, respectively, the RICT Program will not be voluntarily entered. However, it is possible that the LCO could be entered for an emergent failure and RICT entry would be allowed.

Discuss an example scenario of a RICT entry for an emergent failure for TS Condition 3.3.1.1.A, 3.3.1.1.B, or 3.3.6.1.

1 Polickoski, J.T., TVA, letter to the NRC, "License Amendment Request to Revise Technical Specifications to Adopt Risk-Informed Completion Times" dated March 31, 2022 (ML22090A287)

Question 11 (STSB) – Justification for ECCS Instrumentation Variation

Attachments 2.1, 2.2, and 2.3 to the LAR contain the proposed TS markups for Browns Ferry, Units 1, 2, and 3, respectively.[1] In TS 3.3.5.1, "ECCS Instrumentation," each of the Required Actions proposed for the RICT program includes a loss of function in Conditions B, C, E, F, and G. Per TSTF-505, Revision 2, loss of function is a prohibited condition. Provide a justification for this variation.

1 Polickoski, J.T., TVA, letter to the NRC, "License Amendment Request to Revise Technical Specifications to Adopt Risk-Informed Completion Times" dated March 31, 2022 (ML22090A287)

Question 12 (STSB) – Apparent Loss of Function for TS Conditions

Table E1-1 of Enclosure 1 to the LAR includes descriptions of the design success criteria (DSC) of TSs proposed for inclusion in the RICT program.[1] Given the inoperable equipment in the TS condition, the DSC represents the minimum set of remaining credited equipment sufficient to perform TS safety function while in an RICT. The following TS conditions appear to include a loss of function. Clarify the associated DSCs in Table E1-1:

a) TS 3.3.5.1.E (One or more channels inoperable for Function 3.f, HPCI Pump Discharge Flow – Low (Bypass))

b) TS 3.5.1.C (HPCI System inoperable)

c) TS 3.5.1.D (HPCI System inoperable and Condition A entered)

d) TS 3.5.3.A (RCIC System inoperable)

e) TS 3.8.1 E (Two required offsite circuits inoperable)

1 Polickoski, J.T., TVA, letter to the NRC, "License Amendment Request to Revise Technical Specifications to Adopt Risk-Informed Completion Times" dated March 31, 2022 (ML22090A287)