Tennessee Valley Authority, 1101 Market Street, Chattanooga, Tennessee 37402

CNL-19-065

July 15, 2019

10 CFR 50.69
10 CFR 50.90

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

Watts Bar Nuclear Plant, Units 1 and 2
Facility Operating License Nos. NPF-90 and NPF-96
NRC Docket Nos. 50-390, 50-391, and 72-1048

SUBJECT: **Partial Response to NRC Request for Additional Information Regarding Watts Bar Nuclear Plant, Units 1 and 2, Application to Adopt 10 CFR 50.69, "Risk-informed Categorization and Treatment of Structures, Systems, and Components for Nuclear Power Reactors" (WBN-TS-17-24) (EPID L-2018-LLA-0493)**

References:

1. TVA letter to NRC, CNL-18-068, "Watts Bar Nuclear Plant, Units 1 and 2, Application to Adopt 10 CFR 50.69, 'Risk-informed Categorization and Treatment of Structures, Systems, and Components for Nuclear Power Reactors' (WBN-TS-17-24)," dated November 29, 2018 (ML1834A363)

2. NRC Electronic Mail to TVA, "Watts Bar Nuclear Plant - Final Request for Additional Information Related to Application to Adopt 10 CFR 50.69 (EPID L-2018-LLA-0493)," dated June 18, 2019 (ML19169A359)

In Reference 1, Tennessee Valley Authority (TVA) submitted for Nuclear Regulatory Commission (NRC) approval, a request for an amendment to Facility Operating License Nos. NFP-90 and NPF-96 for the Watts Bar Nuclear Plant (WBN), Units 1 and 2, to modify the WBN Facility Operating Licenses to allow for the implementation of the provisions of Title 10 of the *Code of Federal Regulations* (10 CFR) Part 50.69, "Risk-Informed Categorization and Treatment of Structures, Systems and Components for Nuclear Power Reactors." In Reference 2, the NRC submitted a request for additional information (RAI) and requested that TVA respond by July 15, 2019, for those responses to questions not requiring uncertainty estimates and by July 29, 2019, for those responses to questions requiring uncertainty estimates.

Accordingly, Enclosure 1 to this letter provides the TVA responses to NRC DRA RAIs-01, 02, 06, 07, 09, and 11 of Reference 2. As noted in Reference 2, the remaining RAI responses will be provided to the NRC by July 29, 2019. As noted in Enclosure 1, the TVA response to DRA RAI-01 requires a revision to the proposed WBN Units 1 and 2 Facility Operating Licenses that were provided in Reference 1. Enclosure 2 to this letter provides the existing WBN Unit 1 and Unit 2 Facility Operating Licenses marked-up to show the proposed changes. Enclosure 3 to this letter provides the existing WBN Unit 1 and Unit 2 Facility Operating Licenses re-typed pages to show the proposed changes. Enclosures 2 and 3 supersede those Operating License changes provided in Reference 1.

The enclosures to this letter do not change the no significant hazards consideration nor the environmental considerations contained in Reference 1. Additionally, in accordance with 10 CFR 50.91 (b)(1), TVA is sending a copy of this letter and the enclosures to the Tennessee Department of Environment and Conservation.

There are no new regulatory commitments made in this letter. Please address any questions regarding this submittal to Kimberly D. Hulvey at (423) 751-3275.

I declare under penalty of perjury that the foregoing is true and correct. Executed on the 15th day of July 2019.

Respectfully,

Signing For:

James T. Polickoski
Interim Director, Nuclear Regulatory Affairs

Enclosures:

1. Partial Response to NRC Request for Additional Information Regarding Watts Bar Nuclear Plant, Units 1 and 2, Application to Adopt 10 CFR 50.69, "Risk-informed Categorization and Treatment of Structures, Systems, and Components for Nuclear Power Reactors" (WBN-TS-17-24) (EPID L-2018-LLA-0493)
2. WBN Units 1 and 2 Facility Operating Licenses Changes Markup
3. WBN Units 1 and 2 Facility Operating Licenses Changes Retyped Copy

cc: NRC Regional Administrator - Region II
NRC Senior Resident Inspector - Watts Bar Nuclear Plant
NRC Project Manager - Watts Bar Nuclear Plant
Division of Radiological Health - Tennessee State Department of Environment and Conservation

Partial Response to NRC Request for Additional Information Regarding Watts Bar Nuclear Plant, Units 1 and 2, Application to Adopt 10 CFR 50.69, "Risk-informed Categorization and Treatment of Structures, Systems, and Components for Nuclear Power Reactors" (WBN-TS-17-24) (EPID L-2018-LLA-0493)

NRC Introduction

Title 10 of the Code of Federal Regulations, Section 50.69 (10 CFR 50.69), "Risk-Informed Categorization and Treatment of Structures, Systems, and Components for Nuclear Power Reactors", allows licensees to use a risk-informed process to categorize systems, structures, and components (SSCs) according to their safety significance in order to remove SSCs of low safety significance from the scope of certain identified special treatment requirements. Regulatory Guide (RG) 1.201, Revision 1, "Guidelines for Categorizing Structures, Systems, and Components in Nuclear Power Plants According to their Safety Significance" (Agencywide Documents Access and Management System (ADAMS) Accession No. ML061090627) endorses, with regulatory positions and clarifications, the Nuclear Energy Institute (NEI) guidance document NEI 00-04, Revision 0 "10 CFR 50.69 SSC Categorization Guideline", (ADAMS Accession No. ML052910035) as one acceptable method for use in complying with the requirements in 10 CFR 50.69. Both RG 1.201 and NEI 00-04 cite RG 1.200, "An Approach for Determining the Technical Adequacy of Probabilistic Risk Assessment Results for Risk-Informed Activities," (ADAMS Accession No. ML040630078) which endorses industry consensus probabilistic risk assessment (PRA) standards, as the basis against which peer reviews evaluate the technical adequacy of a PRA. Revision 2 of RG 1.200 is available at ADAMS Accession No. ML090410014.

By letter dated November 29, 2018 (ADAMS Accession No. ML18334A363), Tennessee Valley Authority (TVA), submitted a license amendment request (LAR) to adopt 10 CFR 50.69, Risk-informed Categorization and Treatment of Structures, Systems, and Components for Watts Bar Nuclear Plant (WBN), Units 1 & 2. Section 3.1.1 of the LAR states that TVA will implement the risk categorization process in accordance with NEI 00-04, Revision 0, as endorsed by RG 1.201. However, TVA's LAR does not contain sufficient information for the Nuclear Regulatory Commission (NRC) staff to determine whether TVA has implemented the guidance in NEI 00-04, as endorsed by RG 1.201, appropriately to demonstrate compliance with all the requirements in 10 CFR 50.69. The following requests for additional information (RAIs) outline the specific issues and information needed to complete the NRC staff's review:

NRC DRA RAI 01 – Appendix X, Close-out of Facts and Observations (APLA)

Section 2 of RG 1.200 states for the applicable technical requirements, "the staff anticipates that current good practice, i.e., Capability Category II (CC II) of the American Society of Mechanical Engineers (ASME)/American Nuclear Society (ANS) PRA standard, is the level of detail that is adequate for the majority of applications," and that a peer review is needed to determine if the intent of the requirements in the standard is met. The primary result of a peer review are the Facts and Observations (F&Os) recorded by the peer review team. The process to close *finding-level F&Os is documented in Appendix X to NEI 05-04, 07-12, and 12-13 "Close-out of Facts and Observations (F&Os)[1]," as accepted by NRC in letter dated May 3, 2017[2].*

Section 3.3 of the LAR states that a finding closure review was conducted on the internal events (including internal floods) PRA (IEPRA) model in June 2017 and for the seismic PRA (SPRA) in April 2017.

a. Provide the following information to confirm that the F&O closure review for internal events, including internal flooding, was performed consistent with Appendix X to NEI 05-04, 07-12, and 12-13, as accepted by the staff, with conditions.

   i. Confirm that the Independent Assessment team was provided with and performed an independent written assessment that included justification of whether the resolution for each F&O constituted a PRA upgrade or maintenance update, as defined in the ASME/ANS ASME/ANS RA-Sa-2009 PRA Standard and endorsed by RG 1.200, Revision 2.

   OR

   ii. Alternatively, perform a subsequent Independent Assessment for F&O(s) closure and/or addendum to the Independent Assessment report to address the inconsistency with Appendix X, as accepted, with conditions, by the NRC staff via letter dated May 3, 2017. Provide any F&Os or items remaining open as a result of this review. For each F&O and/or item that remains open, provide its associated disposition to demonstrate that it has no adverse impact on the 10 CFR 50.69 risk-informed application.

b. Appendix X guidance states in part, [t]he relevant PRA documentation should be complete and have been incorporated into the PRA model and supporting documentation prior to closing the finding. For closure after the on-site review, Appendix X guidance further states, "[t]he host utility may, in the time between the on-site review and the finalization of the Independent Assessment team report, demonstrate that the issue has been addressed, that a closed finding has been achieved, and that the documentation has been formally incorporated in the PRA Model of Record [MOR]."

   Attachment 2 of the LAR states, the internal events (including internal flooding) PRA (IEPRA) models to be used in categorization for Units 1 and 2 are both Revision 3, dated March 2017 for the respective MORs. The NRC staff notes that the F&O closure review for the internal events, including internal flooding, PRA occurred in June 2017.

   i. Confirm that all model changes associated with the closure of all F&Os reviewed during the Independent Assessment performed in June 2017 for the IEPRA (includes internal floods) model were incorporated into the PRA models and/or the supporting documentation at the time of the finalization of the Independent

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

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

Assessment team report, consistent with the staff's acceptance and conditions provided in the letter dated May 3, 2017.

OR

ii. *Perform a subsequent Independent Assessment for F&O closure and/or addendum to the Independent Assessment F&O closure report to address the identified inconsistency with Appendix X, as accepted, with conditions, by the NRC staff in letter dated May 3, 2017. Provide any F&Os that remain open as a result of this review. For each F&O and/or item that remains open, provide its associated disposition to demonstrate that it has no adverse impact on the 10 CFR 50.69 risk-informed application.*

OR[3]

iii. *Alternatively, propose a mechanism that assures all the PRA model and documentation changes reviewed by the Independent Assessment team for the closure of all F&Os in the final Independent Assessment report are incorporated into the MOR(s) prior to implementation of the 10 CFR 50.69 risk-informed categorization.*

c. *Appendix X guidance states in part, "[i]n some cases, the Independent Assessment team may be assembled such that some reviewers are only needed for a limited number of finding reviews, and it may be possible to have these reviewers participate remotely. This remote participation should be supported with web and teleconference connection to the on-site review team, and the remote reviewers should participate in relevant consensus sessions."*

   i. *If remote (i.e. subsequent reviews) were performed following the Independent Assessment team's onsite review, describe the scope of the remote review performed. Include details for the NRC staff to confirm consistency with Appendix X (i.e., if the subsequent review and consensus session was remote using web conferencing, or face-to-face and the number of participants).*

   OR

   ii. *Alternatively, perform a subsequent Independent Assessment for F&O closure and/or addendum to the Independent Assessment report to address the identified inconsistency with Appendix X, as accepted, with conditions, by the NRC staff in letter dated May 3, 2017. Provide any F&Os that remain open as a result of this review. For each F&O and/or item that remains open, provide its associated disposition to demonstrate it has no adverse impact on the 10 CFR 50.69 risk-informed application.*

d. *Appendix X guidance states in part, the team will review the Supporting Requirement (SR) to ensure that the aspects of the underlying SR that were previously not met, or met at [Capability Category] CC I, are now met, or met at CC II.*

   i. *Explain how closure of all F&Os was assessed to ensure that the capabilities of the PRA elements, or portions of the PRA within the elements, associated with the closed F&Os now meet ASME/ANS RA-Sa-2009 SRs at CC II.*

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[3] NRC Electronic Mail to TVA, "Watts Bar Nuclear Plant - Correction to Final Request for Additional Information Related to Application to Adopt 10 CFR 50.69 (EPID L-2018-LLA-0493)," dated July 5, 2019

OR

ii. Alternatively, perform a subsequent Independent Assessment for F&O closure and/or addendum to the Independent Assessment report to address the inconsistency with the Appendix X process, as accepted, with conditions by the NRC staff in letter dated May 3, 2017. Provide any F&Os that remain open as a result of this review. For each F&O and/or item that remains open, provide its associated disposition to demonstrate that it has no adverse impact on the 10 CFR 50.69 risk-informed application.

TVA Response to NRC DRA RAI 01a

The Independent Assessment (IA) team was provided descriptions of how each finding level of Facts & Observations (F&O) was resolved prior to the on-site review. However, the information provided did not include a written assessment and justification of whether the resolution of each F&O, within the scope of the IA, constitutes a PRA upgrade or maintenance update as required by NEI 05-04 Appendix X Section X.1.3. The absence of this update/upgrade self assessment did not negatively impact the ability of the IA team in performance of their review because the team based their conclusions on the merits for each F&O resolution. The team's assessment of whether the resolution constituted a PRA upgrade or maintenance update is based on consensus of the Independent Assessment team. None of the changes made to the PRA were considered by the team to constitute a PRA upgrade or the use of a new method.

TVA Response to NRC DRA RAI 01b

DRAI 01b allows the option to perform items b.i, or b.ii, or b.iii. TVA is responding to option b.iii as follows.

The IA F&O Closure Report provides the closure evaluation for each F&O. Confirmation of modeling changes or associated documents are noted in the acceptance evaluation column. TVA is proposing a license condition that requires the MOR to be updated with the F&O resolutions from the closed F&Os prior to system categorization (see next to last bullet in Attachment 1 to this enclosure).

Because of the addition of this new action to Attachment 1, the following action that was in Attachment 1 in the referenced letter is no longer needed:

- Findings and Observations (F&Os) items 2-28 and 7-10. The issues associated with these F&Os will be corrected.

Enclosure 2 contains the markup of the proposed license condition reflecting the above deletion and Enclosure 3 contains the retyped proposed license condition.

Reference

TVA letter to NRC, CNL-18-068, "Watts Bar Nuclear Plant, Units 1 and 2, Application to Adopt 10 CFR 50.69, 'Risk-informed Categorization and Treatment of Structures, Systems, and Components for Nuclear Power Reactors' (WBN-TS-17-24)," dated November 29, 2018 (ML1834A363)

TVA Response to NRC DRA RAI 01c

DRA 01c allows the option to perform items c.i, or c.ii. TVA is responding to option c.i as follows.

The IA team members participated on-site at TVA; therefore, there were no remote reviewers. TVA provided the IA team with information for seven F&Os that were initially assessed to be open during the on-site review. These subsequent reviews and consensus sessions were supported by use of WebEx and/or teleconference dependent on the complexity of the subject under review by the same three-person team that performed the on-site review.

TVA Response to NRC DRA RAI 01d

The IA team reviewed each F&O to ensure the resolution used to close the F&O met the requirements of each associated ASME/ANS RA-Sa-2009 Supporting Requirement at Capability Category II.

NRC DRA RAI 02 – Seismic PRA Peer-Review and Use of Appendix X, Close-out of Facts and Observations for the Seismic PRA (RILIT)

Section 2.2 of RG 1.200, Revision 2, provides regulatory guidance regarding peer reviews and the staff regulatory position on NEI 00-02, "Probabilistic Risk Assessment (PRA) Peer Review Process Guidance" (ADAMS Accession No. ML061510619), 05-04 "Process for Performing Follow-On [Internal Events] PRA Peer Reviews Using the American Society of Mechanical Engineers (ASME)/American Nuclear Society (ANS) PRA Standard" (ADAMS Accession No. ML083430462), and 07-12 "Fire Probabilistic Risk Assessment (FPRA) Peer Review Process Guidelines" (ADAMS Accession No. ML102230070). Section 2 of TVA's response to the 10 CFR 50.54(f) information request arising from Near Term Task Force (NTTF) recommendation 2.1 (ADAMS Accession No. ML17181A485) states that the seismic PRA (SPRA) peer review was performed in accordance with the guidance in NEI 12-13. NRC letter, "U.S. Nuclear Regulatory Commission Acceptance of Nuclear Energy Institute (NEI) Guidance NEI 12-13, "External Hazards PRA Peer Review Process Guidelines," (August 2012)," dated March 7, 2018 (ADAMS Accession No. ML18025C025), provides the staff clarifications and qualifications on this guidance for seismic and external hazard PRA peer reviews. Further, the staff accepted the F&O independent assessment process, with conditions, in NRC letter, "U.S. Nuclear Regulatory Commission Staff Expectations For An Industry Facts And Observations Independent Assessment Process," dated May 3, 2017 (ADAMS Accession No. ML17079A427). The LAR does not discuss the consideration of the staff's clarifications and qualifications on NEI 12-13 during the performance of the peer review for the licensee's SPRA or the consideration of the staff's conditions on the F&O independent assessment process used for closure of the SPRA finding level F&Os.

Discuss how the SPRA peer review and the F&O independent assessment considered the staff's clarifications and qualifications on NEI 12-13 in the NRC letter dated March 7, 2018, and the staff's conditions on the use of the F&O independent assessment process in the NRC letter dated May 3, 2017. Provide justification for not considering specific clarifications, qualifications, or conditions in those letters in the context of this application.

TVA Response to NRC DRA RAI 02

The response to RAI APLB-01 in the referenced letter also applies to the response to NRC DRA RAI 02 for the WBN 50.69 LAR.

Reference

TVA letter to NRC, CNL-19-035, "Response to Request for Additional Information Regarding Application for Technical Specification Change Regarding Risk-Informed Justification for the Relocation of Specific Surveillance Frequency Requirements to a Licensee Controlled Program (WBN-TS-18-14) (EPID L-2018-LLA-0279)," dated March 21, 2019 (ML19127A323)

NRC DRA RAI 06 – Alternate Method Proposed to Assess Contribution from Internal Fires (APLA)

Paragraph (c)(1)(ii) of 10 CFR 50.69 requires that the licensee determine the SSC's functional importance using an integrated, systematic process for addressing initiating events (internal and external), SSCs, and plant operating modes, including those not modeled in the plant specific PRA.

Section 3.2.2 of the LAR states in part, "[t]he WBN categorization process will use the Fire Safe Shutdown Equipment List (SSEL) for evaluation of safety significance related to fire hazards." It further states that this approach addresses conditions defined by 10 CFR 50, Appendix R, NRC Branch Technical Position CMEB 9.5-1, regulatory exemptions, and fire-induced multiple spurious operations to identify equipment. The LAR describes this approach as an alternate process from the NEI 00-04 endorsed approaches and is considered to be a conservative method, compared to the FIVE methodology or fire PRA, based on industry assessments.

Section 3.3 of NEI 00-04, Revision 0, provides limited guidance for determining the technical adequacy attributes required for these types of analyses for this specific application. RG 1.201, Revision 0, states in part, "as part of the plant-specific application requesting to implement §50.69, the licensee or applicant will provide the bases supporting the technical adequacy of its…non-PRA-type analyses for this application."

Address the following regarding the proposed alternate approach:

a. *Provide justification that use of the Fire SSEL method is technically adequate relative to the acceptable methods identified in NEI 00-04. Include in the justification: (1) the industry assessments referenced in the LAR and (2) TVA summary of the industry evaluations and how the results from the evaluations support the conclusion that the TVA's proposed approach to use the Fire SSEL is conservative. The justification provided should also demonstrate how additional SSCs will be assigned high safety significance (HSS) with TVA's approach compared to using a previously accepted method (e.g., additional SSCs would not be identified in a FIVE or fire PRA analysis when compared to the Fire SSEL method).*
b. *Section 3.2.2 of the LAR states in part, "[t]he fire safe shutdown paths identify the safety functions and associated sets of equipment credited to achieve and maintain safe shutdown under postulated fire conditions" and that, "[t]he fire SSEL identifies the credited equipment." Section 3.2.2 of the LAR also states "additional equipment that is relied upon to establish and maintain safe shutdown will be retained as HSS." In review of Figure 3.1*

of the LAR, it appears there are other SSCs, not on the Fire SSEL, that may be considered for safe shutdown. According to Figure 3.1, if an SSC is not already on the SSEL, the next step in the process is to question whether the SSC is relied upon to maintain safe shutdown for a fire. An affirmative response to this question would categorize the SSC as candidate HSS.

i. *Provide clarification along with a rationale for the additional equipment that will be identified as HSS for a fire event that is not on the SSEL.*

ii. *Confirm that all the SSCs identified as candidate HSS per Figure 3.1 of the LAR will remain HSS at the end of the categorization and cannot be recategorized by the Integrated Decision-making Panel.*

c. *Clarify whether the fire detection and suppression (and fire dampers) equipment is included on WBNs SSEL. If not included, summarize how the risk-significance of this equipment will be evaluated to determine whether the equipment is HSS or low safety significance (LSS).*

d. *Fire protection actions can be credited if they are "feasible and reliable" but PRA actions generally are not credited unless they are proceduralized and have a failure probability assigned. Provide discussion for how the probability of failure of operator actions is incorporated/considered in the analysis for determining SSCs identified on the SSEL.*

TVA Response to NRC DRA RAI 06

The response to RAI 07 in the referenced letter also applies to the response to NRC DRA RAI 06 for the WBN 50.69 LAR.

Reference

TVA letter to NRC, CNL-19-002, "Response to Request for Additional Information Regarding Application to Modify Sequoyah Nuclear Plant Units 1 and 2, Application to Adopt 10 CFR50.69, 'Risk-informed Categorization and Treatment of Structures, Systems, and Components for Nuclear Power Reactors,' (SQN-TS-17-06) (EPID: L-2018-LLA-0066)," dated March 21, 2019 (ML19081A065)

NRC DRA RAI 07 - Integrated PRA Hazards Model (APLA/RILIT)

Paragraph (c)(1)(ii) of 10 CFR 50.69 requires that the SSC functional importance be determined using an integrated, systematic process. NEI 00-04, Section 5.6, "Integral Assessment," discusses the need for an integrated computation using available importance measures. It further states that the "integrated importance measure essentially weighs the importance from each risk contributor (e.g., internal events, fire, seismic PRAs) by the fraction of the total core damage frequency [or large early release frequency] contributed by that contributor." The guidance provides formulas to compute the integrated Fussel-Vesely (FV), and integrated Risk Achievement Worth (RAW).

Based on the information provided in the LAR, it is not clear to NRC staff how TVA proposes to address the integration of importance measures across all hazards (i.e., internal events, internal flooding, and seismic). Considering these observations provide the following:

a. *Explain how the integration of importance measures across hazards for the 10 CFR 50.69 categorization process will be performed and whether it will be performed using an integrated one-top (single top gate) model across multiple PRA hazards.*

b. *Discuss how the individual importance measures (e.g., FV and RAW) for the PRA model are derived and justify why the importance measures generated do not deviate from the NEI guidance or Table 3-1 of the LAR. If the practice or method used to generate the integrated importance measures is determined to deviate from the NEI guidance, then provide justification to support why the integrated importance measures computed are appropriate for use in the categorization process.*

c. *Describe how the importance measures (i.e., FV and RAW) for the PRA one-top, all hazards model are derived for the SPRA considering that the seismic hazard is discretized into 'bins.' The discussion should include how the same basic events, which were discretized by binning during the development of the SPRA, are then combined (i.e., combined across 'bins' as well as across failure modes such as seismic and random failure modes) to develop representative importance measures. Further, discuss how they are compared to the importance measure thresholds in NEI 00-04. Provide justification to support the determined impact on the categorization results and describe how the approach is consistent with the guidance in NEI 00-04.*

d. *In the context of the "integral assessment" described in Section 5.6 of NEI 00-04, it is understood that importance evaluations performed in accordance with the process in NEI 00-04 are determined on a component basis. However, it is not apparent from the LAR and the NEI 00-04 guidance how the integrated importance measures are calculated for certain components where corresponding basic events, which represent different failure modes for a component, in the SPRA may not align with basic events in other PRA modeled hazards. Examples of such basic events include those that are specific to the SPRA, including implicitly modeled components, or basic events that represent a subcomponent modeled within the boundary of an internal events PRA component.*

   Provide details and justification to support how the integrated importance measures will be calculated for the SPRA modeled basic events that may not align directly with basic events modeled in other PRA hazards. Include discussion for any 'mapping' that will be performed across the SPRA basic events and those in other PRA modeled hazards where additional modelling is determined to be necessary.

<u>TVA Response to NRC DRA RAI 07a</u>

As noted in Section 3.1.1 of the referenced letter, "The process to categorize each system will be consistent with the regulatory endorsed guidance in NEI 00-04." The approach described in Section 1.5 of NEI 00-04 regarding "Integrated Importance Assessment" states, "In order to facilitate an overall assessment of the risk significance of SSCs, an integrated computation is performed using the available importance measures. This integrated importance measure essentially creates a weighted-average importance based on the importance measures and the risk contributed by each hazard (e.g., internal events, fire, seismic PRAs)." Therefore, a one-top model will not be utilized to assess the importance of an SSC and instead a weighted average using the importance measures and corresponding CDF/LERF from each of the hazards will be used as described in NEI 00-04.

<u>Reference</u>

TVA letter to NRC, CNL-18-068, "Watts Bar Nuclear Plant, Units 1 and 2, Application to Adopt 10 CFR 50.69, 'Risk-informed Categorization and Treatment of Structures, Systems, and Components for Nuclear Power Reactors' (WBN-TS-17-24)," dated November 29, 2018 (ML1834A363)

TVA Response to NRC DRA RAI 07b

As noted in the response to NRC DRA RAI 07a, TVA is following the NEI 00-04 process regarding individual importance measures; therefore, the process used is consistent with Table 3-1 of the referenced letter. The process followed does not deviate from the industry guidance (NEI 04-10).

Reference

TVA letter to NRC, CNL-18-068, "Watts Bar Nuclear Plant, Units 1 and 2, Application to Adopt 10 CFR 50.69, 'Risk-informed Categorization and Treatment of Structures, Systems, and Components for Nuclear Power Reactors' (WBN-TS-17-24)," dated November 29, 2018 (ML1834A363)

TVA Response to NRC DRA RAI 07c

As noted in the response to NRC DRA RAI 07a, TVA will not assess any importance measures based on a PRA one-top all hazards model and instead will take each of the importance measures and perform a weighted average based on its CDF/LERF weighting as described in Section 1.5 regarding "Integrated Importance Assessment" and Section 5.6, "Integral Assessment," of NEI 00-04.

TVA Response to NRC DRA RAI 07d

As noted in NRC's DRA RAI 07d, the process employed by TVA is justified because the importance evaluations are performed in accordance with NEI 00-04, which are determined on a component basis. Some components in the Internal Events PRA may not be modeled explicitly in the Seismic PRA due to it being subsumed in a super-component (as described in NEI 00-04), or because it was screened out of the analysis. For those that were screened out, the importance measure would be taken as non-risk significant (0 for Fussell-Vessely or 1 for Risk Achievement Worth). For the components that are included under a super component, the super component risk importance would instead be considered to determine risk significance.

DRA RAI 09 – Addition of FLEX to the PRA Model (APLA/RILIT)

The NRC memorandum dated May 30, 2017, "Assessment of The Nuclear Energy Institute 16-06, 'Crediting Mitigating Strategies in Risk-Informed Decision Making,' Guidance for Risk-Informed Changes to Plants Licensing Basis" (ADAMS Accession No. ML17031A269), provides the NRC's staff assessment of challenges to incorporating FLEX equipment and strategies into a PRA model in support of risk-informed decision making in accordance with the guidance of RG 1.200. The LAR does not state whether or not TVA has incorporated FLEX mitigating strategies and associated equipment into the PRA models used to support this application. Therefore, it is unclear whether FLEX equipment and operator actions are modeled in the PRA models used to support this application and, if applicable, whether the incorporation of FLEX equipment and actions into the PRA models was performed in an acceptable manner. Provide the following information separately for internal events PRA, SPRA, and external hazard screening as appropriate:

a. *Clarify whether FLEX equipment and associated actions have been credited in the PRAs used to support this application, identifying the specific PRA(s) that include such credit. If not incorporated or their inclusion is not expected to impact the PRA results used in the categorization process, no response to parts (b) and (c) is requested.*

b. *If the FLEX equipment or operator actions have been credited, and their inclusion is expected to impact the PRA results used in the categorization process, provide the following information separately for the IEPRA (includes internal floods) and SPRA, as appropriate:*

   i. *A discussion detailing the extent of incorporation, i.e. summarize the supplemental equipment and compensatory actions that have been quantitatively credited for each of the PRA models used to support this application.*

   ii. *If any credited FLEX equipment is dissimilar to other plant equipment credited in the PRA (i.e. SSCs with sufficient plant-specific or generic industry data), discuss the data and failure probabilities used to support the modeling and provide the rationale for using the chosen data. Include discussion on whether the uncertainties associated with the parameter values are in accordance with the ASME/ANS PRA Standard as endorsed by RG 1.200, Revision 2.*

   iii. *If any operator actions related to FLEX equipment are evaluated using approaches that are not consistent with the endorsed ASME/ANS RA-Sa-2009 PRA Standard (e.g., using surrogates), discuss the methodology used to assess human error probabilities for these operator actions. The discussion should include:*

      1. *A summary of how the impact of the plant-specific human error probabilities and associated scenario-specific performance shaping factors listed in (a)-(j) of supporting requirement HR-G3 of the ASME/ANS RA-Sa-2009 PRA Standard were evaluated.*

      2. *Whether maintenance procedures for the portable equipment were reviewed for possible pre-initiator human failures that renders the equipment unavailable during an event, and if the probabilities of the pre-initiator human failure events were assessed as described in HLR-HR-D of the ASME/ANS RA-Sa-2009 PRA standard.*

      3. *If the procedures governing the initiation or entry into mitigating strategies are ambiguous, vague, or not explicit, a discussion detailing the technical bases for probability of failure to initiate mitigating strategies.*

c. *The ASME/ANS RA-Sa-2009 PRA standard defines PRA upgrade as the incorporation into a PRA model of a new methodology or significant changes in scope or capability that impact the significant accident sequences or the significant accident progression sequences. Section 1-5 of Part 1 of ASME/ANS RA-Sa-2009 states that upgrades of a PRA shall receive a peer review in accordance with the requirements specified in the peer review section of each respective part of this Standard.*

   i. *Provide an evaluation of the model changes associated with incorporating non-safety related SSCs that were included following the FLEX mitigation strategies (permanently installed and/or portable) but are not similar to safety-related SSCs, which demonstrates that none of the following criteria is satisfied: (1) use of new methodology, (2) change in scope that impacts the significant accident sequences or the significant accident progression sequences, (3) change in capability that* *impacts the significant accident sequences or the significant accident progression sequences,*

OR

ii. Propose a mechanism to ensure that a focused-scope peer review is performed on the model changes associated with incorporating mitigating strategies, and associated F&Os are resolved to Capability Category II prior to implementation of the 10 CFR 50.69 categorization program.

TVA Response to NRC DRA RAI 09a

The response to RAI APLB-05a in the referenced letter also applies to the response to NRC DRA RAI 09a for the WBN 50.69 LAR, with the exception that the WBN PRA model also screened external hazards.

Reference

TVA letter to NRC, CNL-19-035, "Response to Request for Additional Information Regarding Application for Technical Specification Change Regarding Risk-Informed Justification for the Relocation of Specific Surveillance Frequency Requirements to a Licensee Controlled Program (WBN-TS-18-14) (EPID L-2018-LLA-0279)," dated March 21, 2019 (ML19127A323)

TVA Response to NRC DRA RAI 09b

i. The response to RAI APLB-05a in the referenced letter addressed the NRC memorandum dated May 30, 2017, and noted that the WBN internal events with internal flooding PRA model development followed the guidance of the ASME/ANS PRA standard in crediting the permanently installed FLEX diesel generators (DGs) and supporting components.

ii. As noted in the response to RAI APLB-05a in the referenced letter:

> "WBN does not include portable FLEX equipment in the PRA models. WBN includes the permanently installed FLEX DGs within the PRA model and supporting components including fuel tank, alignment of breakers, buses, and operator actions to align the FLEX DGs. The failure probabilities for the equipment was assumed to be the same as other components of the same type already included within the model (e.g. the FLEX DGs have the same failure probabilities as the Emergency Diesel Generators)."

A method to provide the justification for use of safety related equipment failure probabilities is to perform a sensitivity study that increases the failure probability for modeled FLEX equipment. A sensitivity study was performed whereby the FLEX failure probabilities for the basic events were increased by a factor of three. The internal events impact based on the sensitivity resulted in no change to the importance measures when compared to the base model importance measures.

Furthermore, the sensitivity case performed for the SPRA model determined that component importance risk rankings for the SPRA do not introduce unique events compared to the internal events results for a multiplication of three in the base FLEX component random failure probabilities.

iii. The response to RAI APLB-05b in the referenced letter addresses the WBN procedures regarding the mitigating strategies for use of the permanently installed FLEX DGs and the use of the HRA Calculator to address the human performance shaping factors for each of the FLEX Human Error Probabilities (HEPs). As noted in response to RAI APLB-05b, "Each type of Diesel Generator has a separate HEP, as the method to start and align the diesels are different. Both of these HEPs were developed in accordance with the corresponding technical elements in the NRC endorsed ASME/ANS PRA Standard." The standard referred to in the quoted sentence is the ASME/ANS RA-Sa-2009 PRA Standard.

<u>Reference</u>

TVA letter to NRC, CNL-19-035, "Response to Request for Additional Information Regarding Application for Technical Specification Change Regarding Risk-Informed Justification for the Relocation of Specific Surveillance Frequency Requirements to a Licensee Controlled Program (WBN-TS-18-14) (EPID L-2018-LLA-0279)," dated March 21, 2019 (ML19127A323)

<u>TVA Response to NRC DRA RAI 09c</u>

NRC DRA 09c allows the option to respond to items i or ii. TVA is responding to item ii as noted below.

Consistent with the definition of PRA Upgrade from the ASME/ANS RA-Sa-2009 standard, the inclusion of the permanently installed FLEX diesel generators and supporting equipment does not represent a new methodology nor does it represent a significant change in scope or capability that impacts the significant accident sequences or the significant accident progression sequences. Therefore, inclusion of the FLEX diesels and supporting equipment in both the internal events and the seismic PRA models does not constitute an upgrade to the PRA models. Therefore, a focus-scoped peer review would not be required consistent with the ASME/ANS RA-Sa-2009 standard.

<u>NRC DRA RAI 11 – Use of Addendum B of the PRA Standard (2013) (RILIT)</u>

Section 4 of RG 1.200, Revision 2, states that a risk informed submittal should contain discussions concerning peer review. If the peer review is not performed against the established standards, then information needs to be included in the submittal demonstrating that the different criteria used are consistent with the established standards, as endorsed by NRC.

Section 3.2.3 of Enclosure 1 to the LAR states that the seismic PRA was peer reviewed against the requirements in the ASME/ANS PRA Standard (ASME/ANS RA-Sb-2013). RG 1.200, Revision 2, endorses ASME/ANS PRA Standard Addendum A (ASME/ANS RA-Sa-2009). As noted in the NRC letter dated July 6, 2011, "U.S. Nuclear Regulatory Commission (NRC) Comments on "Addenda to a Current ANS: ASME RA-SB - 20XX, Standard for Level 1/Large Early Release Frequency Probabilistic Risk Assessment for Nuclear Power Plant Applications" (ADAMS Accession No. ML111720067), NRC did not endorse Addendum B of the PRA Standard. TVA's seismic PRA peer review was performed using a PRA Standard different from that endorsed by the NRC staff in RG 1.200, Revision 2.

Discuss how the supporting requirements (SRs) in Addendum B, which is not endorsed by the NRC for licensing applications, and the NRC staff's comments in the above cited letter dated July 6, 2011, are consistent with the SRs in Part 5 of Addendum A, for this application. If the different criteria are not consistent with the endorsed Standard, describe how the analogous Addendum A supporting requirements have been met.

TVA Response to NRC DRA RAI 11

The response to RAI APLB-02 in the referenced letter also applies to the response to NRC DRA RAI 11 for the WBN 50.69 LAR.

Reference

TVA letter to NRC, CNL-19-035, "Response to Request for Additional Information Regarding Application for Technical Specification Change Regarding Risk-Informed Justification for the Relocation of Specific Surveillance Frequency Requirements to a Licensee Controlled Program (WBN-TS-18-14) (EPID L-2018-LLA-0279)," dated March 21, 2019 (ML19127A323)

Attachment 1: List of Categorization Prerequisites

TVA will establish procedure(s) prior to the use of the categorization process on a plant system. The procedure(s) will contain the elements/steps listed below.

- IDP member qualification requirements.

- Qualitative assessment of system functions. System functions are qualitatively categorized as preliminary HSS or LSS based on the seven criteria in Section 9 of NEI 00-04 (see Section 3.2 of this enclosure). Any component supporting an HSS function is categorized as preliminary HSS. Components supporting, an LSS function are categorized as preliminary LSS.

- Component safety significance assessment. Safety significance of active components is assessed through a combination of PRA and non-PRA methods, covering all hazards. Safety significance of passive components is assessed using a methodology for passive components.

- Assessment of DID and safety margin. Safety-related components that are categorized as preliminary LSS are evaluated for their role in providing DID and safety margin and, if appropriate, upgraded to HSS.

- Review by the IDP. The categorization results are presented to the IDP for review and approval. The IDP reviews the categorization results and makes the final determination on the safety significance of system functions and components.

- Risk sensitivity study. For PRA-modeled components, an overall risk sensitivity study is used to confirm that the population of preliminary LSS components results in acceptably small increases to CDF and LERF and meets the guidelines of Regulatory Guide 1.174.

- Periodic reviews are performed to ensure continued categorization validity and acceptable performance for those SSCs that have been categorized.

- Documentation requirements per Section 3.1.1 of the enclosure.

In addition to the procedure changes above, TVA will also perform the following actions.

- As documented in the F&O Closure Report, all changes initiated by the F&O resolutions were confirmed by the Integrated Assessment Team to have been incorporated into the living model and associated documentation. TVA shall update the Model of Record (MOR) with this information prior to system categorization.
- TVA shall re-introduce the State of Knowledge Correlation (SOKC) into the MOR prior to using the PRA model to support categorization of SSCs under 10 CFR 50.69.
- With respect to the external flooding hazards, TVA shall re-confirm that there is sufficient time to eliminate the source of the threat or to provide an adequate response in accordance with screening criterion C5, prior to 50.69 categorization.

Enclosure 2

WBN Units 1 and 2 Facility Operating Licenses Changes Markup

(10) By May 31, 2018, TVA shall ensure that a listing organization acceptable to the NRC (as the Authority Having Jurisdiction) determines that the fire detection monitoring panel in the main control room either meets the appropriate designated standards or has been tested and found suitable for the specified purpose.

(11) The licensee shall replace the WBN, Unit 1 upper compartment cooler cooling coils with safety-related cooling coils to eliminate a potential source of containment sump dilution during design basis events prior to increasing the number of Tritium Producing Burnable Absorber Rods (TPBARs) loaded in the reactor core above 704.

(12) Adoption of 10 CFR 50.69, “Risk-Informed categorization and treatment of structures, systems and components for nuclear power plants”

(a) TVA is approved to implement 10 CFR 50.69 using the processes for categorization of Risk- Informed Safety Class (RISC)-1, RISC-2, RISC- 3, and RISC-4 structures, systems, and components (SSCs) using: Probabilistic Risk Assessment (PRA) models to evaluate risk associated with internal events, including internal flooding, and seismic hazards; the shutdown safety assessment process to assess shutdown risk; the Arkansas Nuclear One, Unit 2 (ANO-2) passive categorization method to assess passive component risk for Class 2 and Class 3 SSCs and their associated supports; the results of non-PRA evaluations that are based on the IPEEE Screening Assessment for External Hazards; fire hazards by use of the fire protection program (FPP) safe shutdown equipment list (SSEL), and a screening of other external hazards updated using the external hazard screening significance process identified in ASME/ANS PRA Standard RA-Sa-2009, as specified in Unit 1 License Amendment [Number].

(b) Prior to implementation of the provisions of 10 CFR 50.69, TVA shall complete the implementation items in Enclosure 1, Attachment 1, "WBN 10 CFR 50.69 PRA Implementation Items," to TVA letter CNL-19-065, “Partial Response to NRC Request for Additional Information Regarding Watts Bar Nuclear Plant, Units 1 and 2, Application to Adopt 10 CFR 50.69, ‘Risk-informed Categorization and Treatment of Structures, Systems, and Components for Nuclear Power Reactors” (WBN-TS-17-24) (EPID L-2018-LLA-0493)’,” dated July 15, 2019.

(13) Prior NRC approval, under 10 CFR 50.90, is required for a change to the categorization process specified above (e.g., change from using the FPP SSEL approach to an internal fire probabilistic risk assessment approach).

D. The following exemptions are authorized by law, will not present an undue risk to the public health and safety, and are consistent with the common defense and security. Therefore, these exemptions are granted pursuant to 10 CFR 50.12.

(1) Deleted

TVA may make changes to the approved fire protection program without prior approval of the Commission, only if those changes would not adversely affect the ability to achieve and maintain safe shutdown in the event of a fire.

(9) By May 31, 2018, TVA shall report that a listing organization acceptable to the NRC (as the Authority Having Jurisdiction) has determined that the fire detection monitoring panel in the main control room either meets the appropriate designated standards or has been tested and found suitable for the specified purpose.

(10) TVA will verify for each core reload that the actions taken if $F_Q^W(Z)$ is not within limits will assure that the limits on core power peaking $F_Q(Z)$ remain below the initial total peaking factor assumed in the accident analyses.

(11) TVA will implement the compensatory measures described in Section 3.4, “Additional Compensatory Measures,” of TVA Letter CNL-18-012, dated January 17, 2018, during the timeframe the temperature indicator for RCS hot let 3 is not required to be operable for the remainder of Cycle 2. If the RCS hot leg 3 temperature indicator is returned to operable status prior to the end of Cycle 2, then these compensatory measures are no longer required.

(12) Adoption of 10 CFR 50.69, “Risk-Informed categorization and treatment of structures, systems and components for nuclear power plants”

(a) TVA is approved to implement 10 CFR 50.69 using the processes for categorization of Risk- Informed Safety Class (RISC)-1, RISC-2, RISC- 3, and RISC-4 structures, systems, and components (SSCs) using: Probabilistic Risk Assessment (PRA) models to evaluate risk associated with internal events, including internal flooding, and seismic hazards; the shutdown safety assessment process to assess shutdown risk; the Arkansas Nuclear One, Unit 2 (ANO-2) passive categorization method to assess passive component risk for Class 2 and Class 3 SSCs and their associated supports; the results of non-PRA evaluations that are based on the IPEEE Screening Assessment for External Hazards; fire hazards by use of the fire protection program (FPP) safe shutdown equipment list (SSEL), and a screening of other external hazards updated using the external hazard screening significance process identified in ASME/ANS PRA Standard RA-Sa-2009, as specified in Unit 2 License Amendment [Number].

(b) Prior to implementation of the provisions of 10 CFR 50.69, TVA shall complete the implementation items in Enclosure 1, Attachment 1, "WBN 10 CFR 50.69 PRA Implementation Items," to TVA letter CNL-19-065, “Partial Response to NRC Request for Additional Information Regarding Watts Bar Nuclear Plant, Units 1 and 2, Application to Adopt 10 CFR 50.69, ‘Risk-informed Categorization and Treatment of Structures, Systems, and Components for Nuclear Power Reactors” (WBN-TS-17-24) (EPID L-2018-LLA-0493)’,” dated July 15, 2019.

(13) Prior NRC approval, under 10 CFR 50.90, is required for a change to the categorization process specified above (e.g., change from using the FPP SSEL approach to an internal fire probabilistic risk assessment approach).

D. The licensee shall have and maintain financial protection of such types and in such amounts as the Commission shall require in accordance with Section 170 of the Atomic Energy Act of 1954, as amended, to cover public liability claims.

E. This license is effective as of the date of issuance and shall expire at midnight on October 21, 2055.

FOR THE NUCLEAR REGULATORY COMMISSION

William M. Dean, Director
Office of Nuclear Reactor Regulation

Appendices:
1. Appendix A – Technical Specifications
2. Appendix B – Environmental Protection Plan

Date of Issuance: October 22, 2015

Enclosure 3

WBN Units 1 and 2 Facility Operating Licenses Changes Retyped Copy

(10) By May 31, 2018, TVA shall ensure that a listing organization acceptable to the NRC (as the Authority Having Jurisdiction) determines that the fire detection monitoring panel in the main control room either meets the appropriate designated standards or has been tested and found suitable for the specified purpose.

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(b) Prior to implementation of the provisions of 10 CFR 50.69, TVA shall complete the implementation items in Enclosure 1, Attachment 1, "WBN 10 CFR 50.69 PRA Implementation Items," to TVA letter CNL-19-065, "Partial Response to NRC Request for Additional Information Regarding Watts Bar Nuclear Plant, Units 1 and 2, Application to Adopt 10 CFR 50.69, 'Risk-informed Categorization and Treatment of Structures, Systems, and Components for Nuclear Power Reactors" (WBN-TS-17-24) (EPID L-2018-LLA-0493)'," dated July 15, 2019.

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FOR THE NUCLEAR REGULATORY COMMISSION

William M. Dean, Director
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Appendices:
1. Appendix A –
   Technical Specifications
2. Appendix B –
   Environmental Protection Plan

Date of Issuance: October 22, 2015