TENNESSEE VALLEY AUTHORITY
1101 Market Street, Chattanooga, Tennessee 37402

CNL-21-033

April 28, 2021

10 CFR 50.90

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

Browns Ferry Nuclear Power Plant Units 1, 2, and 3
Renewed Facility Operating License Nos. DPR-33, DPR-52, and DPR-68
NRC Dockets 50-259, 50-260, and 50-296

Subject: **Response to Request for Additional Information Regarding License Amendment Request to Adopt 10 CFR 50.69, "Risk-Informed Categorization and Treatment of Structures, Systems and Components for Nuclear Power Reactors" (EPID L-2020-LLA-0162)**

References:
1. TVA Letter to NRC, CNL-20-002, "Browns Ferry Nuclear Plant, Units 1, 2, and 3, Application to Adopt 10 CFR 50.69, 'Risk-Informed Categorization and Treatment of Structures, Systems and Components for Nuclear Power Reactors' (TS-BFN-529)," dated July 17, 2020 (ML20199M373)

2. NRC Electronic Mail to TVA, "Browns Ferry Nuclear Plant, Units 1, 2, and 3 - Request for Additional Information Regarding Request to Adopt 10 CFR 50.69 (EPID L-2020-LLA-0162)," dated February 10, 2021 (ML21041A542)

In Reference 1, Tennessee Valley Authority (TVA) submitted a request for an amendment to modify the Browns Ferry Nuclear Plant (BFN) licensing basis, by the addition of a License Condition, to allow for the implementation of the provisions of Title 10 of the *Code of Federal Regulations*, Part 50.69, "Risk-Informed Categorization and Treatment of Structures, Systems and Components for Nuclear Power Reactors." In Reference 2, the Nuclear Regulatory Commission (NRC) provided a request for additional information (RAI) and requested a response by March 29, 2021, which was subsequently extended to April 28, 2021. Enclosure 1 provides the TVA RAI response. As provided for in the RAI response, TVA proposes additional activities for Attachment 1 of Reference 1 (List of Categorization Prerequisites):

- Establish limitations before interfacing structures, systems, and components can be categorized without categorizing the entire interfacing system.
- Document that all the open Internal Events Probabilistic Risk Assessment and Fire Probabilistic Risk Assessment (Fire PRA) Facts and Observations have been closed using an NRC-accepted Appendix X Independent Assessment process.

- The resolutions to the internal events findings with the potential to impact the Fire PRA modeling will be incorporated into the Fire PRA. Following the model updates, the values of total Core Damage Frequency (CDF) and total Large Early Release Frequency (LERF) for each unit will be evaluated for conformance to the Regulatory Guide 1.174 risk acceptance guidance. The SOKC will be evaluated for importance by assessing the mean risk results relative to acceptance guidelines.
- Ensure resolutions to the internal events findings with the potential to affect the Seismic PRA modeling will be incorporated into the Seismic PRA.
- Assess the impact on the internal events with internal flooding "living model" with respect to the risk importance measures used to assign the safety classification (high or low) from pending model changes to be compared to previously categorized system SSCs to confirm that the criteria for Low Safety Significance and High Safety Significance is still applicable, and reclassify, if necessary, in accordance with Nuclear Energy Institute (NEI) 00-04 (i.e., PRA model update, and at least once per two fuel cycles in a unit).

Additionally, TVA noted that the first paragraph of the Reference 1 List of Categorization Prerequisites alluded to modifications needed to achieve an overall CDF and LERF consistent with NRC Regulatory Guide 1.174 risk limits. There are no outstanding plant modifications required, so this paragraph has been deleted.

A revised List of Categorization Prerequisites is provided in Enclosure 2 that reflects these new activities and changes. Enclosure 3 provides an update to the Reference 1 License Condition to cite Enclosure 2 of this letter.

The enclosures to this letter do not change the no significant hazards consideration or 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 Alabama State Department of Public Health.

There are no new regulatory commitments contained in this submittal. If you should have any questions regarding this submittal, please contact Kimberly Hulvey, Senior Manager, Fleet Licensing, at 423-751-3275.

I declare under penalty of perjury that the foregoing is true and correct. Executed on this 28th day of April 2021.

Respectfully,

James T. Polickoski
Director, Nuclear Regulatory Affairs

Enclosures:

cc: See Page 3

Enclosures:

1. Response to Request for Additional Information Regarding License Amendment Request to Adopt 10 CFR 50.69, “Risk-Informed Categorization and Treatment of Structure, Systems and Components for Nuclear Power Reactors”

2. List of Categorization Prerequisites

3. BFN Units 1, 2, and 3 Renewed Facility Operating License Markup Pages

cc (Enclosures):

NRC Regional Administrator - Region II
NRC Senior Resident Inspector - Browns Ferry Nuclear Plant
NRC Project Manager – Browns Ferry Nuclear Plant
State Health Officer, Alabama State Department of Public Health

Enclosure 1

Response to Request for Additional Information Regarding License Amendment Request to Adopt 10 CFR 50.69, "Risk-Informed Categorization and Treatment of Structure, Systems and Components for Nuclear Power Reactors" (EPID L-2020-LLA-0162)

The Nuclear Regulatory Commission (NRC) Request for Additional Information provided the following background discussion:

By letter dated July 17, 2020 (Agencywide Documents Access and Management System (ADAMS) Accession No. ML20199M373), the Tennessee Valley Authority (TVA, the licensee) submitted a license amendment request (LAR) for Browns Ferry Nuclear Plant, Units 1, 2, and 3 (Browns Ferry). The proposed amendments would allow for the voluntary adoption of the requirements of Title 10 of the Code of Federal Regulations (10 CFR), Section 50.69, "Risk-informed categorization and treatment of structures, systems and components for nuclear power reactors," at Browns Ferry. The NRC staff is reviewing the LAR and has identified where additional information is needed to complete its review. The NRC staff's request for additional information (RAI) is below.

The NRC RAI questions are provided in italics. The TVA responses are provided in non-italicized font.

RAI 01 – Open PRA Facts and Observations (F&O)

Regulatory Guide (RG) 1.200, Revision 2 (ADAMS Accession No. ML090410014) provides guidance for addressing probabilistic risk assessment (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 facts and observations (F&Os) recorded by the peer review team and the subsequent resolution of these F&Os. 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)" (ADAMS Package Accession No. ML17086A431), which was accepted by the NRC in a letter dated May 3, 2017 (ADAMS Accession No. ML17079A427), describes a process to close finding-level F&Os.

LAR Attachment 3 presents all findings not closed by the F&O closure review for both internal events and fire PRA (IEPRA and FPRA). The NRC staff reviewed the open findings and associated dispositions and has identified that the licensee's stated disposition for each F&O is to resolve the finding "and have it closed in accordance with an Independent Assessment." The NRC staff notes that the last item in the Categorization Prerequisite List regards a commitment to "close all open F&Os listed in [LAR] Attachment 3 and incorporate changes into the MOR [model of record] prior to system categorization." Thus, the license condition proposed in LAR Enclosure 2 does not appear to commit to close the open findings using an Independent Assessment, as stated in the F&O dispositions. In addition, the NRC staff notes that the resolutions to many of the internal events F&Os appear to have the potential to impact the FPRA. However, the LAR does not indicate that the FPRA will be updated to incorporate the resolutions to the internal events findings. Additionally, the NRC staff observed the TVA's F&O

Closure Review conducted December 7-10, 2020, which appeared to close many of the F&Os provided in the LAR.

In light of these observations, address the following:

a) *Provide any necessary updates to the status of open F&Os.*

b) *Confirm that the license condition is meant to commit to resolving all the open IEPRA and FPRA F&Os and to closing them with an Appendix X Independent Assessment process, as accepted by the NRC in letter dated May 3, 2017, and, as applicable, update the wording in the license condition or in LAR Attachment 1, accordingly.*

c) *Alternatively, if the intention is not to commit to resolving the open IEPRA and FPRA F&Os and closing them with an Independent Assessment, then update the dispositions for the open F&Os and provide justification that the existing PRA modeling associated with the open F&Os does not impact this application.*

d) *Confirm that the resolutions to the internal events findings with the potential to impact the FPRA modeling will be incorporated into the FPRA.*

e) *If it is confirmed in the response to part (d), above, that the resolutions to the internal events findings with the potential to impact the FPRA modeling will be incorporated into the FPRA, then propose a mechanism that ensures that the internal events findings with the potential to impact the FPRA modeling will be incorporated into the FPRA prior to the implementation of the risk categorization program.*

f) *If it is not confirmed in the response to part (d), above, that the resolutions to the internal events findings with the potential to impact the FPRA modeling will be incorporated into the FPRA, then justify that incorporation of the IEPRA finding resolutions into the FPRA would have no impact on the risk categorization program.*

TVA Response to RAI 01:

<u>Response a</u>

The Browns Ferry Nuclear Plant (BFN) IEPRA and FPRA Finding Level F&O Closure Review by Independent Assessment was held virtually via Microsoft Teams from December 7th through 10th, 2020. The purpose was to perform an independent assessment in accordance with Appendix X of NEI 05-04, NEI 07-12, and NEI 12-13 to review close out of “Finding” level F&Os of record from prior PRA peer reviews against the PRA Standard. As described in the RAI background material above, following Appendix X to these NEI documents was found by the NRC to be an acceptable process to close finding level F&Os.

The BFN F&O Closure by Independent Assessment Team consisted of five team members, with extensive qualifications and average of over 24 years of experience covering all areas of IEPRA and FPRA. All reviewers met the criteria specified in NEI 05-04, NEI 07-12, NEI 12-13, and the ASME/ANS PRA Standard.

Following the virtual on-site review held from December 7th through 10th, 2020, the plant modeling team addressed several of the initial findings and these F&Os were reassessed by the review team with a second consensus session held on January 26, 2021. Following this

reassessment, all finding level F&Os included in LAR Attachment 3 were assessed as fully closed by the independent assessment team.

Response b

TVA confirms the meaning of the license condition, as stated by this RAI. As discussed in the response to RAI 1(a), all the open IEPRA and FPRA F&Os have been closed with an NRC-accepted Appendix X Independent Assessment process. As described in the RAI background material above, following Appendix X to these NEI documents was found by the NRC to be an acceptable process to close finding level F&Os. The List of Categorization Prerequisites referenced in the BFN 10 CFR 50.69 LAR License Condition is updated as follows:

> All the open IEPRA and FPRA F&Os have been closed using an NRC-accepted Appendix X Independent Assessment process.

Refer to Enclosures 2 and 3 of this submittal.

Response c

As described in the responses to RAI 1(a) and 1(b), all the open IEPRA and FPRA F&Os have been closed with an NRC-accepted Appendix X Independent Assessment process. Therefore, this question is not applicable.

Response d

The resolutions to the internal events findings with the potential to impact the FPRA modeling will be incorporated into the FPRA prior to implementation of the Categorization Process. The mechanism to ensure this is described in the response to RAI 4(e).

Response e

As stated in Response RAI 1(d) the resolutions to the internal events findings with the potential to impact the FPRA modeling will be incorporated into the FPRA prior to implementation of the Categorization Process. The List of Categorization Prerequisites referenced in the BFN 10 CFR 50.69 LAR License Condition is updated as follows:

> The resolutions to the internal events findings with the potential to impact the FPRA modeling will be incorporated into the FPRA.

Refer to Enclosures 2 and 3 of this submittal.

Response f

Based on the response to RAI 1(d), this question is not applicable.

RAI 02 – Identification of Key Assumptions and Sources of Uncertainties

Paragraphs (c)(1)(i) and (ii) of 10 CFR 50.69 require that a licensee's PRA be of sufficient quality and level of detail to support the structure, system, and component (SSC) categorization process, and that all aspects of the integrated, systematic process used to characterize SSC

importance must reasonably reflect the current plant configuration and operating practices, and applicable plant and industry operational experience.

Section 5 of NEI 00-04, Revision 0, "10 CFR 50.69 SSC Categorization Guideline" (ADAMS Accession No. ML052900163), provides guidance for performing sensitivity studies for each PRA model to address the uncertainty associated with those models. Specifically, Sections 5.1, 5.2, and 5.3 provide guidance for such sensitivities for the internal events, fire and seismic PRA, respectively. The sensitivity studies are performed to ensure that assumptions and sources of uncertainty (e.g., human error, common cause failure, and maintenance probabilities) do not mask importance of components.

LAR Section 3.1.10 explains that TVA used the detailed process of identifying, characterizing and qualitative screening of model uncertainties found in Section 5.3 of NUREG-1855, Revision 1, "Guidance on the Treatment of Uncertainties Associated with PRAs in Risk-Informed Decision Making" (ADAMS Accession No. ML17062A466), and Section 3.1.1 of Electric Power Research Institute (EPRI) Technical Report (TR) 1016737, "Treatment of Parameter and Modeling Uncertainty for Probabilistic Risk Assessments." LAR Attachment 6 presents a total of six FPRA key assumptions and sources of uncertainty and LAR Section 3.1.10 explains that no internal events or seismic PRA modeling uncertainties key to this application were identified. The LAR states that a "list of assumptions and sources of uncertainty" were reviewed to identify those which would be significant to this application and that if a "non-conservative treatment" or a method "not commonly accepted" were used then it was reviewed for its impact in application. The LAR does not explain how the initial "list of assumptions and sources of uncertainty" was developed nor does it indicate whether plant-specific issues, generic industry concerns, and modeling choice concerns (e.g., level of detail) were all reviewed to compile this list. It is also not clear to the NRC staff whether other screening criteria beyond identification of non-conservative treatments and uncommon practices were used to screen sources of uncertainty.

Section 3.2.3 of RG 1.200, Revision 2, as well as NUREG-1855, Revision 1, provide guidance on how to identify, characterize, and treat key sources of uncertainty relevant to a risk-informed application. Revision 1 of NUREG-1855 additionally cites EPRI TR 1026511, "Practical Guidance on the Use of Probabilistic Risk Assessment in Risk-Informed Applications with a Focus on the Treatment of Uncertainty." Furthermore, Section 1.3 of NUREG-1855, Revision 1, states, in part, that "[a]lthough assumptions and approximations made on the level of detail in a PRA can influence the decision-making process, they are generally not considered to be model uncertainties because the level of detail in the PRA model could be enhanced, if necessary. Therefore, methods for identifying and characterizing issues associated with level of detail are not explicitly included in NUREG-1855; they are, however, addressed in EPRI TRs 1016737 and 1026511." Additionally, Section 3.3.2 of RG 1.200, Revision 2, defines key assumptions and sources of uncertainty. Therefore, the NRC staff requests the following information to confirm that the key assumptions and sources of uncertainty provided in Attachment 6 of the LAR were properly assessed from the base PRAs that have received peer reviews:

a) *Provide a description of the process used to determine the key sources of uncertainty and assumptions for each PRA model used to support this application. The discussion should be provided separately for the IEPRA, FPRA, and seismic PRA (SPRA) and include:*

    i. *A description of how the key assumptions and sources of uncertainties provided in Attachment 6 were identified from the initial comprehensive list of PRA model(s) (i.e.,*

base model) source of uncertainties and assumptions, including those associated with plant-specific features, modeling choices, and generic industry concerns. This can include an identification of the sources of plant-specific and applicable generic modeling uncertainties identified in the uncertainty analyses for the base internal events and internal flooding PRA.

ii. Discussion and justification that the evaluation criteria used to identify an assumption or source of uncertainty as "key" is consistent with RG 1.200, and/or NUREG-1855, Revision 1, Revision 2, or other NRC-accepted method.

b) If the process of identifying "key" assumptions or sources of uncertainty for the PRA models used to support this application cannot be justified for use in the 10 CFR 50.69 categorization process, provide the results of an updated assessment that includes a description of each key assumption or source of uncertainty identified.

TVA Response to RAI 02

a) TVA performed a review of all assumptions and uncertainties (and how they were developed), and have judged that none meet the criteria to be "key" to the 10 CFR 50.69 application. The following sub-bullets describe the process used for each PRA model: Internal Events, Fire, and Seismic.

i. TVA reevaluated all sources of uncertainty and assumptions in a manner that was much more comprehensive than performed previously. TVA started with the Assumptions and Uncertainty Notebook prepared in response to a similar RAI issued for the Surveillance Frequency Control Program (reference RAI APLB/C-03 [ML20356A106]). A comprehensive list of assumptions and sources of uncertainty was compiled, including those associated with plant-specific features, modeling choices, and generic industry concerns. A disposition was developed for each assumption and source of uncertainty, addressing the impact on the 10 CFR 50.69 application. The risk metrics of interest for are CDF and LERF due to internal events (including internal flooding and fire) and seismic events. For any potential key source of uncertainty or potential key assumption judged not to be key to the application, a discussion was provided to indicate why it does not need to be addressed further in the context of the application.

The BFN PRA documentation review included the notebooks supporting the internal events PRA, internal floods PRA, FPRA, and SPRA. BFN does not have a low power or shutdown PRA, therefore this type of hazard was not considered further.

- Internal Events with Internal Flooding - To identify the key assumptions and uncertainties associated with the Internal Events PRA model supporting the 10 CFR 50.69 application, the generic issues identified in Table A.1 of EPRI 1016737 and EPRI 1026511 were reviewed, as well as the BFN PRA documentation for plant-specific assumptions and uncertainties. The screening criteria applied was taken from NUREG-1855 Revision 1.

- Fire PRA - To identify the key assumptions and uncertainties associated with the PRA models supporting the 10 CFR 50.69 application, the generic issues identified in Table A.1 of EPRI 1016737 and EPRI 1026511 were reviewed, as well as the BFN PRA documentation for plant-specific assumptions and uncertainties. The screening criteria applied was taken from NUREG-1855 Revision 1.

- Seismic PRA - To identify the key assumptions and uncertainties associated with the PRA models supporting the 10 CFR 50.69 application, the generic issues identified in Table A.1 of EPRI 1016737 and EPRI 1026511 were reviewed, as well as the BFN PRA documentation for plant-specific assumptions and uncertainties. The screening criteria applied was taken from NUREG-1855 Revision 1.

ii. Each key assumption and source of uncertainty was evaluated in accordance with NUREG-1855 Revision 1 and RG 1.200 Revision 2 to determine whether it was key for the 10 CFR 50.69 application, using an approach based on screening criteria. A disposition was generated, addressing the impact on the risk metrics of interest for the risk-informed application. The risk metrics of interest for the 10 CFR 50.69 application are core damage frequency (CDF), large early release frequency (LERF), and the risk importance measures (Risk Achievement Worth (RAW) and Fussell-Vesely (FV)).

The screening criteria that were used to determine whether an assumption or source of uncertainty that was key to one or more of the PRA models was key to the application are as follows. If the PRA model key assumption or source of uncertainty satisfied any of these criteria, it was considered to be not key to the application. If the PRA model key assumption or source of uncertainty does not satisfy any of these criteria, it was considered to be key to the application.

- The assumption or source of uncertainty is addressed by implementing a “consensus model” as defined in NUREG-1855 Revision 1. EPRI 1013491 elaborates on the definition of a consensus model to include those areas of the PRA where extensive historical precedent is available to establish a model that has been accepted and yields PRA results that are considered reasonable and realistic. Thus, assumptions and sources of uncertainty for which there is extensive historical precedent, and which produce results that are reasonable and realistic, are screened from further consideration.

- The assumption or source of uncertainty has no impact or insignificant impact on the PRA results and therefore no impact or insignificant impact on the categorization process.

- The assumption or source of uncertainty introduces a realistic conservative bias in the PRA model results. EPRI 1013491 uses the term “realistic conservatisms” and notes that “judiciously applied realistic conservatism can provide a PRA that avoids many of the traps associated with the use of excess conservatism.” This criterion, which allows screening of sources of conservative bias, is intended to be less restrictive than the previous criterion, which does not distinguish between conservative and nonconservative bias. Thus, using this criterion, assumptions that introduce realistic (slight) conservatisms can be screened from further consideration.

- For an assumption under consideration, there is no reasonable alternative assumption or reasonable modeling refinement that would change the risk profile of the plant. Here, “reasonable alternative assumption” is taken with the meaning given in Regulatory Guide 1.200, Revision 2.

- The potential conservatism could result in an overstated risk impact or importance measure, which could influence a decision made based on the 10 CFR 50.69

application. However, the undesired consequence, if any, of overstated risk would be to assign a component a High Safety Significance (HSS), when in reality it could be assigned a Low Safety Significance (LSS). Because retaining the potential conservatism is in that sense fail-safe, the assumption or source of uncertainty is determined to be not key for the application and is not investigated further.

For each BFN identified assumption or source of uncertainty, a qualitative discussion was sufficient to demonstrate that it met one or more of the above screening criteria and therefore, was not key. In some cases, the qualitative evaluation was complemented by insights from PRA results to help screen the assumption or source uncertainty, based on insignificant impact on CDF and LERF, or importance measures.

As a result of this review, it was determined that there are no assumptions or sources of uncertainty that qualify as "key" to the application. As such, the content of Attachment 6 from the LAR is superseded in its entirety with this RAI assessment.

No sensitivity cases were required because there are no key sources of uncertainty relevant to the 10 CFR 50.69 application.

b) This part is not applicable. The process of identifying "key" assumptions or sources of uncertainty for the PRA models used to support the application is justified for use in the 10 CFR 50.69 categorization process.

RAI 03 – Dispositions of Key Assumptions and Sources of Uncertainties

Paragraph (c)(1)(i) of 10 CFR 50.69 requires the licensee to consider the results and insights from the PRA during categorization. The guidance in NEI 00-04 specifies sensitivity studies to be conducted for each PRA model. The sensitivity studies are performed to ensure that assumptions and sources of uncertainty (e.g., human error, common cause failure, and maintenance probabilities) do not mask importance of components. NEI 00-04 guidance states that additional "applicable sensitivity studies" from characterization of PRA adequacy should be considered.

The NRC staff notes that modeling conservatisms (i.e., assumptions and sources of uncertainty) can mask the importance measures of other SSCs. Sections 5.1 and 5.3 of NEI 00-04 provide guidance on performing individual sensitivity studies for key assumptions and sources of uncertainties as part of the categorization process. Section 3.1.10 of the LAR states that "[t]he conclusion of this review is that no additional sensitivity analyses are required to address Browns Ferry PRA model specific assumptions or sources of uncertainty." It is unclear to the NRC staff if any sensitivity studies will be performed for each of the key assumptions and sources of uncertainties provided in Attachment 6 of the LAR and how the determination to either perform or not perform sensitivities was made. Considering these observations, address the following:

a) For any additional key assumptions/sources of uncertainty identified as a result of the response to RAI 02, discuss how each identified key assumption and uncertainty will be dispositioned in the categorization process. The discussion should clarify whether TVA is following the guidance in Section 5 of NEI 00-04 by performing sensitivity analysis or other accepted guidance such as NUREG-1855. The summaries and descriptions should be provided separately for the identified key assumptions and uncertainties related to the IEPRA (includes internal floods), internal FPRA, and SPRA.

LAR Attachment 6 identifies the key assumptions and sources of PRA modeling uncertainty for this application. The NRC staff notes that the LAR presents six key sources of uncertainty for the FPRA and no key sources of uncertainty for the internal events or seismic PRAs. As part of the audit, a comprehensive uncertainty analysis was provided for internal, fire and seismic events that consisted of (1) identification of plant specific assumptions from the PRA notebooks and identification of applicable generic sources of modeling uncertainty from EPRI TRs 1016737 and 1026511, and (2) evaluation and screening of these assumptions and sources of uncertainty to identify key sources of uncertainty. This comprehensive uncertainty analysis was specifically performed for the Browns Ferry TSTF-425 LAR (ADAMS Accession No. ML20087P262). It not clear to the NRC staff whether the conclusions of this analysis are meant to (or do) apply to the 10 CFR 50.69 LAR. The NRC staff notes that the sensitivity of an application to sources of uncertainty can be different for different applications. Therefore, address the following:

b) Clarify whether the uncertainty analysis performed for the TSTF-425 LAR is also the basis for the uncertainty analysis performed for the 10 CFR 50.69. If so, provide justification that disposition of the identified sources uncertainties (especially those identified as "potential key sources of uncertainty") are also applicable to the 10 CFR 50.69 LAR.

c) If the uncertainty analysis performed for the TSTF-425 LAR is not the basis for the 10 CFR 50.69 uncertainty analysis, then describe the uncertainty analysis that was performed for the 10 CFR 50.69 LAR and justify why no key sources of uncertainty were identified for the IEPRA or SPRA.

TVA Response to RAI 03

a) As stated in the RAI-02(a) response, there are no longer any assumptions or sources of uncertainty that qualify as "key." Therefore, no further summaries and descriptions are provided for this response.

b) The uncertainty analysis performed for the TSTF-425 LAR is not the basis for the uncertainty analysis performed for the 10 CFR 50.69 LAR.

c) The BFN 10 CFR 50.69 uncertainty analysis is described in the response to RAI-02(a).

RAI 04 – Total Risk Consideration

Revision 3 of RG 1.174, "An Approach for Using Probabilistic Risk Assessment in Risk-Informed Decisions on Plant-Specific Changes to the Licensing Basis" (ADAMS Accession No. ML17317A2560, provides the risk acceptance guidance in terms of change-in-risk in combination with total core damage frequency defined by regions. These regions are shown in Table 4 and 5 as Region I (No changes allowed), II (Small changes), and III (Very Small Changes and More Flexibility with Respect to Baseline core damage frequency (CDF)/large early release frequency (LERF). NEI 00-04 includes an overall risk sensitivity study for all the Low Safety Significant (LSS) components to assure that if the unreliability of the components was increased, the increase in risk would be small (i.e., meet the acceptance guidelines of RG 1.174, Revision 3).

RG 1.174 and Section 6.4 of NUREG-1855, Revision 1, for a Capability Category II risk evaluation, indicate that the mean values of the risk metrics (total and incremental values) need

to be compared against the risk acceptance guidelines. The mean values referred to are the means of the probability distributions that result from the propagation of the uncertainties on the PRA input parameters and model uncertainties explicitly reelected in the PRA models. In general, the point estimate CDF and LERF obtained by quantification of the cutset probabilities using mean values for each basic event probability does not produce a true mean of the CDF/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).

The NRC staff notes that the LAR does not stipulate whether the total CDF and LERF values presented in LAR Attachment 2 are mean values and notes there is a small margin between the LERF for Units 1 and 2 of 9.3E-06 and 9.4E-06 per year, respectively, and the RG 1.174, Revision 3 LERF threshold of 1E-05 per year. Accordingly, the risk increase due to consideration of the SOKC and the possible risk increase associated IEPRA and FPRA model updates committed to in the license condition to resolve open F&Os could impact the conclusions of the NEI 00-04 Section 8 overall sensitivity study results by increasing the Browns Ferry LERF values above 1E-05 per year.

In light of the observations above, address the following:

a) Demonstrate that for the NEI 00-04 Section 8 overall sensitivity study results, Browns Ferry will be in conformance with the RG 1.174 risk acceptance guidance after the IEPRA and FPRA models are updated to include the increase associated with SOKC (if needed) and potential increases due to committed PRA updates to resolve F&Os.

b) Alternatively, propose a mechanism that ensures that, for NEI 00-04 Section 8 overall sensitivity study results, Browns Ferry will be in conformance with the RG 1.174 risk acceptance guidance after the IEPRA and FPRA models are updated to include the increase associated with SOKC (if needed) and potential increases due to updates to PRA models performed to resolve F&Os.

c) Explain for the update that was performed or will be performed to address the impact of the SOKC which fire PRA parameters are assumed to be correlated beyond the component failure modes frequencies. Include justification that consideration of the identified parameters is sufficient to estimate the impact of the SOKC on fire risk.

TVA Response to RAI 04

a) The FPRA updates have not been completed yet. Therefore, TVA cannot currently demonstrate that the NEI 00-04 Section 8 overall sensitivity study results for BFN will be in conformance with the RG 1.174 risk acceptance criteria. However, the response to RAI 4(b) provides the mechanism that will ensure conformance prior to system categorization.

b) All of the IEPRA and FPRA finding level F&Os have been closed using an NRC accepted Appendix X Independent Assessment process. The existing IEPRA and FPRA models include the SOKC. However, the existing FPRA is based on the IEPRA prior to F&O closure and it is being updated to account for changes to the IEPRA during the F&O closure process. TVA will follow the process outlined in NEI 00-04 and ensure that the NEI 00-04 Section 8 overall sensitivity results will be in conformance with the RG 1.174 risk acceptance guidance following the FPRA model update. The RG 1.174 risk acceptance guidance states that the

total CDF shall be less than 1E-04 per reactor year and that the total LERF shall be less than 1E-05 per reactor year. Following the model update, the values of CDF and LERF for each unit will be evaluated for conformance to the RG 1.174 risk acceptance guidance. This will be tracked through the following item on the List of Categorization Prerequisites provided in the Response to RAI-01(e), as augmented below:

> The resolutions to the internal events findings with the potential to impact the FPRA modeling will be incorporated into the FPRA. Following the model updates, the values of total CDF and total LERF for each unit will be evaluated for conformance to the Regulatory Guide 1.174 risk acceptance guidance. The SOKC will be evaluated for importance by assessing the mean risk results relative to acceptance guidelines.

Refer to Enclosures 2 and 3 of this submittal.

c) The existing FPRA includes SOKC in the model.

The uncertainty analysis assessed the overall uncertainty associated with the BFN FPRA CDF and LERF results. All known uncertainties associated with all contributors to these results were included in the uncertainty propagation, including but not limited to uncertainties in the following parameters:

- Fire ignition frequencies,
- Non-Suppression probabilities,
- Severity Factors
- Hot short probabilities,
- Component failures, and
- Human failures.

(Note: Of those parameters listed above the fire ignition frequencies, non-suppression probabilities, severity factors, and the hot short probabilities are the fire specific parameters. The component failures and human failure events have their uncertainties applied using the same process as the internal events model, through the type codes or basic events themselves).

All fire scenarios that lead to core damage or large early release were included in the uncertainty propagation analysis. This includes scenarios involving control room abandonment and multi-compartment fire analyses.

Uncertainty in the fire ignition frequency, non-suppression probability, and severity factor was combined in order to evaluate the uncertainty in the aggregate fire scenario frequency, as detailed below.

The frequency f of a fire scenario is the sum of the generic ignition source frequencies F_i involved in the fire scenario, weighted by the conditional probability p_i that the ignition source i damages the target set of the fire scenario. This can be summarized with the Equation 1:

$$f = \sum_i F_i \cdot p_i \qquad \text{(Eq. 1)}$$

Each generic ignition source frequency F_i is assigned its uncertainty distribution given in Table 4-4 of NUREG-2169, represented by a lognormal distribution. Each scenario ignition

frequency was calculated as the product of the bin frequency and weighting factor (or the sum of such products). The bin frequency and weighting factor used for each scenario depended on the source assumed for the scenario as defined in Fire Ignition Frequency Task, and the consideration of other parameters such as severity factors and probability of non-suppression defined in the Scoping Fire Modeling and Main Control Room Analysis tasks. This structure allowed the parametric uncertainty analysis to correlate uncertainty in the scenario ignition frequencies with uncertainty in the underlying source bin frequencies (i.e., it accounts for the state of knowledge correlation); thus, this approach avoids the inappropriate assumption that the scenario frequencies are independent of each other.

The probability p_i is evaluated as a point estimate. This point estimate incorporates the contribution of multiple uncertain parameters, including:

- The apportionment fraction of the generic ignition source frequency F_i to the specific ignition source *i* of the scenario under consideration. The uncertainty of this fraction is due to potential inaccuracies in location mapping, equipment counting, and transient weighting factors.

- Fire modeling parameters such as heat release rate, fire growth timeline, non-suppression probability, configuration of ignition source regarding target set, etc.

To represent the uncertainty associated with p_i, a constrained non-informative prior distribution is used. Constrained non-informative prior distributions associated with probability values are discussed in Section 6.3.2.3.3 and on page C-15 of NUREG/CR-6823. This type of prior distribution is selected because it is associated with distributions that are relatively diffuse. As such, it is consistent with a state-of-knowledge where there is relatively good confidence about the value of a point estimate, compared to the uncertainty associated with that point estimate. In this approach, the point estimate is assigned to the mean of the constrained non-informative prior distribution.

Uncertainty in the hot short probabilities (HSPs) is incorporated using a similar approach to the fire ignition frequencies. A type code was developed for each conditional probability of spurious operation and hot short duration conditional probability and then equations were developed for each aggregate HSP applied in the BFN FRANX model. A beta distribution is assigned to each new type code with the variance calculated based on the α and β factors provided in NUREG/CR-7150 and Equation 2 provided below.

$$V = \frac{\alpha \cdot \beta}{(\alpha + \beta)^2 \cdot (\alpha + \beta + 1)} \qquad \text{(Eq. 2)}$$

This was done to correlate uncertainty in the HSPs associated with each fire scenario. This approach avoids the inappropriate assumption that the scenario frequencies are independent of each other.

As mentioned above, the existing FPRA includes SOKC in the model. The fire parameters that include SOKC as well as the justification that they are sufficient to estimate SOKC are also discussed above.

RAI 05 – Categorization of Interfacing SSCs

Section 7.1 of NEI 00-04, Revision 0, "10 CFR 50.69 SSC Categorization Guideline" (ADAMS Accession No. ML052910035), states, "[d]ue to the overlap of functions and components, a significant number of components support multiple functions. In this case, the SSC, or part thereof, should be assigned the highest risk significance for any function that the SSC or part thereof supports." Section 4 of NEI 00-04 states that a candidate low-safety-significant SSC that supports an interfacing system should remain uncategorized until the interfacing system is considered.

The NRC staff notes an apparent inconsistency in LAR Section 3.1.14 concerning treatment of interfacing systems. The sections states that TVA considers its approach to be the same as approved by in the Calvert Cliffs 10 CFR 50.69 Safety Evaluation (SE) (ADAMS Accession No ML19330D909). However, the following partial description of the approach provided in LAR Section 3.1.14 is not consistent with the Calvert Cliffs 10 CFR 50.69 SE:

> *"In some cases, impacts that an interfacing component could have on an interfacing system can be fully determined and the interface component can be categorized (and alternative treatment implemented) without categorizing the entire interfacing system.*
>
> *In this event, an assessment of interface component risk associated with uncategorized systems will be limited to:*
>
> 1. *Cases where an interface component failure cannot prevent performance of interface system functions, or*
>
> 2. *The risk is limited to passive failures assessed as low safety-significant following the passive categorization process for the applicable pressure boundary segments."*

In the Calvert Cliffs 10 CFR 50.69 SE, the NRC-approved approach specifies that both limitations 1 and 2 above should be true before the interfacing SSC can be categorized without categorizing the entire interfacing system.

Therefore, provide a clarification that reconciles this inconsistency regarding TVA's approach to categorizing interfacing SSCs. If TVA's proposed approach differs from that referenced and approved for Calvert Cliffs, provide further explanation and justification for the proposed approach.

TVA Response to RAI 05

TVA will follow the regulatory approved process as described in the Calvert Cliffs Safety Evaluation (ML19330D909) in that both of the following two limitations be true before an interfacing SSC can be categorized without categorizing the entire interfacing system.

1. Cases where an interface component failure cannot prevent performance of interface system functions, and

2. The risk is limited to passive failures assessed as low safety-significant following the passive categorization process for the applicable pressure boundary segments.

Those SSCs can be assessed without performing a full interface system categorization because adequate interface system function knowledge is available to perform the functional assessment and passive risk assessment. Categorizing the entire interfacing system would produce the same functional assessment and passive risk significance for the component. Conformance with the above two limitations for interfacing SSCs has been added to the List of Categorization Prerequisites.

Refer to Enclosures 2 and 3 of this submittal.

RAI 06 – Integrated PRA Hazards Model

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.

Section 5.6 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).

LAR Section 3.3 states that the weighted average importance method presented in NEI 00-04, Section 1.5 will be used to integrate seismic PRA results into the overall importance measures. The licensee cited the response to Watts Bar 10 CFR 50.69 RAI-07 (ADAMS Accession No. ML19196A362) for the integration of risk importance measures across all hazards. The NRC staff notes that SPRA basic events, such as structural failures, may often not align with basic events in other PRA models. The licensee did not mention whether the same approach for Watts Bar 10 CFR 50.69 RAI 07-01 (ADAMS Accession No. ML19302D625) will be applicable to the Browns Ferry 10 CFR 50.69 LAR.

a) Confirm that the response to RAI 07-01 is applicable to Browns Ferry 10 CFR 50.69 LAR.
b) If question a) cannot be confirmed, provide responses to Watts Bar 10 CFR 50.69 RAI 07-01 applicable for Browns Ferry.

TVA Response to RAI 06

a) TVA confirms that the response to RAI 07-01 related to the Watts Bar 10 CFR 50.69 LAR is applicable to the BFN 10 CFR 50.69 LAR.

b) As described above, RAI 6(a) is confirmed. Therefore, RAI 6(b) is not applicable.

RAI 07 – Overall Use of NEI 00-04 Figure 5-6 and Use for External Floods and High Wind

The guidance in NEI 00-04 Figure 5-6 provides guidance to be used to determine SSC safety significance. The guidance in NEI 00-04 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.

LAR Section 3.1.7, states that "[n]o SSCs were explicitly credited to allow a scenario to screen," and therefore, "[s]creened hazards are considered insignificant for every SSC and […] will not be considered during the categorization process." The NRC staff notes that LAR Attachment 4 screens all other external events (besides internal flood, internal fire and seismic events). It appears to NRC staff that, based on this description, 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. The NRC staff notes that plant changes, plant or industry operational experience, 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.

Also, concerning the external flooding, the NRC staff provided an assessment of the Browns Ferry flood hazard mitigating strategy assessment (MSA) dated September 5, 2017 (ADAMS Accession No. ML17222A328). That assessment discusses SSCs that would be relied upon to mitigate the impact of an extreme flooding event such as Local Intense Precipitation (LIP). The report discusses passive features such as external doors but also refers to credit for active components. Section 3.2 of the NRC staff's assessment discusses use of FLEX strategies against external flooding events such as a LIP. Section 3.2.1 of the NRC staff's assessment states that, regarding the Intake Pump Station, each compartment contains sump pumps to remove rainwater that accumulates from openings at the roof. These passive and active SSCs appear to be credited in screening of the external flooding hazard.

In light of these observations, address the following:

a) Clarify whether or not 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) Identify any active and passive SSCs that are credited for screening the external flooding hazard and discuss how those SSCs will be included and considered in the proposed categorization process.

c) Identify any active and passive SSCs that are credited for screening the high winds and tornado hazard, including tornado-generated missiles, and discuss how those SSCs will be included and considered in the proposed categorization process.

TVA Response to RAI 07

a) TVA confirms that the process for evaluating SSCs for other external hazards (i.e., non-PRA modeled hazards) follows the NEI 00-04 Figure 5-6 methodology. For SSCs credited for screened scenarios that if unavailable, would result in an un-screened hazard, the corresponding SSC(s) would be assigned an HSS classification.

b) TVA has evaluated the external flooding hazard for BFN, and has concluded that the only scenario having the potential to affect plant equipment is LIP. With respect to flooding around the lower plant area, the LIP event is postulated to potentially exceed the Current Design Basis LIP Flood of 565 ft. mean sea level (MSL) by as much as 1.6 ft. at the exterior doors leading up to the reactor buildings, intake pumping station, diesel generator buildings, and radwaste building. TVA evaluated the potential water ingress at each of these locations as described below.

Reactor Buildings

The building has an airlock access point for equipment and personnel at the south side of the building that is exceeded by 0.2 ft. by the reevaluated LIP event. The airlock access is a secondary containment boundary that is equipped with inflatable seals in order to maintain an air seal, and the airlock doors are interlocked such that only one door can be opened at once. Given the design considerations, the relatively short duration period of the event and the limited flood height above the door, it is concluded that these buildings will not be jeopardized. Water from a LIP event is also not expected to enter via the north side of the turbine building at the reactor building interface given the existing margin between the probable maximum flood (PMF) used as the design-basis for the area (572.5 ft. MSL) and the reevaluated LIP hazard (566.6 ft. MSL).

The only equipment that is necessary to protect the reactor building from LIP is the reactor building (secondary containment) airlock doors. If the door system was categorized, the airlock doors would be assigned a high safety significant Risk-Informed Safety Class (RISC). SSCs categorized and assigned an HSS classification will be treated as follows: RISC-1, no change in existing special treatment requirements, RISC-2 SSCs will be evaluated to determine if additional treatments are necessary to ensure reliability.

Intake Pumping Station

This building has a floor elevation of 564.7 ft. MSL and access curbs located at the entrance doors with elevation 565.2 ft. MSL. In total, four doors exist that correspond to each one of the four residual heat removal service water (RHRSW) pump compartments. These watertight external doors are normally closed and are designed to withstand a PMF of 578 ft. MSL; therefore, no LIP runoff is expected to enter the building compartments. With regards to openings at the roof that would allow the entry of water, each compartment contains two sump pumps that would remove rainwater. Furthermore, a single sump pump is capable of removing the rain with coincident RHRSW pump seal failure and emergency equipment cooling water strainer leakage. As a result, it is concluded that this building will not be negatively impacted by the LIP event.

The intake pumping station is protected from LIP by the watertight access doors. If the doors are categorized, they would be assigned an HSS RISC. If the sump system were to be categorized, this equipment would be assigned an HSS RISC. SSCs categorized and assigned an HSS classification will be treated as follows: RISC-1, no change in existing special treatment requirements, RISC-2 SSCs will be evaluated to determine if additional treatments are necessary to ensure reliability.

Diesel Generator Buildings

The buildings have a floor elevation of 565.5 ft. MSL and are exceeded by the reevaluated LIP hazard by up to 1.1 ft. There are five watertight exterior doors which are normally closed and are designed to withstand the design-basis PMF water elevation of 578 ft. MSL. Given the design considerations and the relatively short duration period of the event, it is concluded that these buildings will not be negatively impacted by the LIP event.

The diesel generator buildings are protected from LIP by watertight access doors. If the doors are categorized, they would be assigned an HSS RISC. SSCs categorized and assigned an HSS classification will be treated as follows: RISC-1, no change in existing

special treatment requirements, RISC-2 SSCs will be evaluated to determine if additional treatments are necessary to ensure reliability.

Radwaste Building

This building has a floor elevation of 565 ft. MSL that would be exceeded by the reevaluated LIP hazard by approximately 1.2 ft. at three exterior doors. The exterior doors are watertight and are designed to withstand the design-basis PMF water elevation of 578 ft. MSL. In addition, the equipment in the radwaste building is not considered essential to maintaining the reactors in a safe configuration.

The three radwaste building exterior doors are protected from LIP by watertight seals. If the doors are categorized, they would be assigned a high safety significant RISC. SSCs categorized and assigned an HSS classification will be treated as follows: RISC-1, no change in existing special treatment requirements, RISC-2 SSCs will be evaluated to determine if additional treatments are necessary to ensure reliability.

Summary

The following SSCs are credited to mitigate LIP, and therefore are assigned a HSS classification, hence, RISC-1 or RISC-2, if categorized.

- Reactor Building Airlock Doors
- Diesel Generator Building Watertight Doors
- Intake Pumping Station Watertight Doors
- Intake Pumping Station Rain Water Sump Pumps
- Radwaste Building Watertight Doors

Additionally, FLEX equipment is outside the scope of SSCs eligible to be categorized under 10 CFR 50.69 at Browns Ferry.

c) There were no active or passive SSCs identified in the screening of the high winds or tornado hazards. However, TVA's categorization process requires the assessment of a full scope of hazards, including other external risks (e.g., tornadoes, external floods, etc.). This ensures that during the categorization process of a given system, all SSCs in that system are assessed for other external hazards interactions.

RAI 08 – Propagation of Closed and Open/Partially Open Findings from Internal Events Open Finding Level F&O

According to Section 5-1.2 of the 2009 ASME/ANS PRA Standard it is assumed that full-scope internal-events at-power Level 1 and Level 2 LERF PRAs exist and that those PRAs are used as the basis for the SPRA. Therefore, the acceptability of the IEPRA model used as the foundation for the SPRA is an important consideration. Section 3.2 of the LAR states that the internal events and seismic hazards findings were reviewed and closed using the process documented in Appendix X to NEI 05-04, NEI 07-12, and NEI 12-13. Further, Attachment 3 of the LAR provides an evaluation of internal events open finding level F&Os that impact the 10 CFR 50.69 applications. However, the LAR does not provide information about the propagation of changes made to the IEPRA (includes internal floods) for (1) resolving the finding level F&Os that are closed, and (2) addressing the open/partially open finding level F&Os. The

NRC staff's concerns in this RAI are similar to those discussed in Watts Bar 10 CFR 50.69 RAI 12-01 (ADAMS Accession No. ML19302D625).

a. *Clarify whether changes made to the internal events model to close finding level F&Os or to disposition the open/partially open finding level F&Os that are applicable to the SPRA, have been implemented in the SPRA used to support this application or justify not implementing the changes in the context of impact on this application.*

b. *Describe an approach that is consistent with the requirements in 10 CFR 50.69(e) and the guidance in NEI 00-04 for appropriate categorization of SSCs to propagate changes in the IEPRA (includes internal floods) to the SPRA arising from the review of this application, as part of any implementation item resulting from this application, or as part of routine maintenance and updating of the IEPRA (includes internal floods).*

TVA Response to RAI 08

Response a

Changes made to the internal events model, which includes flooding, to close finding level F&Os that are applicable to the SPRA will be evaluated and implemented in the BFN SPRA prior to implementing this application. The List of Categorization Prerequisites referenced in the BFN 10 CFR 50.69 LAR License Condition is updated as follows:

> The resolutions to the internal events findings with the potential to affect the SPRA modeling will be incorporated into the SPRA.

Refer to Enclosures 2 and 3 of this submittal.

Response b

TVA maintains a 'living model' that assesses the change in risk due to changes in the as-built, as-operated plant. In order to establish an approach that will appropriately propagate changes in the Internal Events PRA and Internal Flooding PRA to the SPRA, the List of Categorization Prerequisites referenced in the BFN 10 CFR 50.69 LAR License Condition is updated as follows:

> TVA will assess the impact on the internal events with internal flooding "living model" with respect to the risk importance measures used to assign the safety classification (high or low) from pending model changes to be compared to previously categorized system SSCs to confirm that the criteria for LSS and HSS is still applicable, and reclassify, in accordance with NEI 00-04, (i.e., PRA model update, and at least once per two fuel cycles in a unit).

Refer to Enclosures 2 and 3 of this submittal.

This approach would be covered by procedure and presented to the Integrated Decision-making Panel for concurrence. As such, this routine periodicity would be independent to the 25% threshold for an off-cycle model update. TVA concludes this approach to be in alignment with 10 CFR 50.69(e) and the guidance of NEI 00-04.

Enclosure 2

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.

- Integrated Decision-Making Panel (IDP) member qualification requirements

- Qualitative assessment of system functions. System functions are qualitatively categorized as preliminary High Safety Significant (HSS) or Low Safety Significant (LSS) based on the seven criteria in Section 9 of Nuclear Energy Institute (NEI) 00-04, "10 CFR 50.69 SSC Categorization Guideline," (see Section 3.2 to Reference 1). 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 Probabilistic Risk Assessment (PRA) and non-PRA methods, covering all hazards. Safety significance of passive components is assessed using a methodology for passive components.

- Assessment of defense-in-depth (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 Core Damage Frequency (CDF) and Large Early Release Frequency (LERF) and meets the acceptance guidelines of Regulatory Guide 1.174.

- Periodic reviews are performed to ensure continued categorization validity and acceptable performance for those structures, systems and components (SSCs) that have been categorized.

- Documentation requirements per Section 3.1.1 of the enclosure to Reference 1.

- As documented in the Facts and Observations (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 will verify the Model of Record (MOR) has been updated with pertinent information to the 10 CFR 50.69 application prior to system categorization.

- The following two limitations will both be true before an interfacing SSC can be categorized without categorizing the entire interfacing system.

    1. Cases where an interface component failure cannot prevent performance of interface system functions, and

2. The risk is limited to passive failures assessed as low safety-significant following the passive categorization process for the applicable pressure boundary segments.

TVA shall close all open F&Os listed in Attachment 3 to Reference 1 and incorporate changes into the MOR prior to system categorization.

- All the open IEPRA and FPRA F&Os have been closed using an NRC-accepted Appendix X Independent Assessment process.

- The resolutions to the internal events findings with the potential to impact the FPRA modeling will be incorporated into the FPRA. Following the model updates, the values of total CDF and total LERF for each unit will be evaluated for conformance to the Regulatory Guide 1.174 risk acceptance guidance. The SOKC will be evaluated for importance by assessing the mean risk results relative to acceptance guidelines.

- The resolutions to the internal events findings with the potential to affect the SPRA modeling will be incorporated into the SPRA.

TVA will assess the impact on the internal events with internal flooding “living model” with respect to the risk importance measures used to assign the safety classification (high or low) from pending model changes to be compared to previously categorized system SSCs to confirm that the criteria for LSS and HSS is still applicable, and reclassify, if necessary, in accordance with NEI 00-04 (i.e., PRA model update, and at least once per two fuel cycles in a unit).

References

1. TVA Letter to NRC, CNL-20-002, “Browns Ferry Nuclear Plant, Units 1, 2, and 3, Application to Adopt 10 CFR 50.69, “Risk-Informed Categorization and Treatment of Structures, Systems and Components for Nuclear Power Reactors” (TS-BFN-529),” dated July 17, 2020 (ML20199M373)

Enclosure 3

BFN Units 1, 2, and 3 Renewed Facility Operating License Markup Pages

Insert 1

(XX) "Adoption of 10 CFR 50.69, "Risk-Informed Categorization and treatment of structures, systems and components for nuclear power plants"

(1) 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, internal fire, and seismic risk; 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; and the results of non-PRA evaluations that are based on the Individual Plant Examination of External Events (IPEEE) Screening Assessment for External Hazards, and a screening of other external hazards updated using the external hazard screening significance process identified in ASME/ANS PRA Standard RA-Sa-2009; Internal fires and seismic hazards are evaluated with BFN specific PRA models; as specified in License Amendment No. [XXX].

(2) TVA shall complete the numbered items listed in Enclosure 2, List of Categorization Prerequisites, of TVA letter [ML Number], dated April 28, 2021, prior to implementation. All issues identified in the enclosure will be addressed and any associated changes will be made, focused scope peer reviews will be performed on changes that are PRA upgrades as defined in the PRA standard (ASME/ANS RA-Sa-2009, as endorsed by RG 1.200, Revision 2), and any findings will be resolved and reflected in the PRA of record prior to implementation of the 10 CFR 50.69 categorization process.

(3) Prior NRC approval, under 10 CFR 50.90, is required for a change to the categorization process specified above (e.g., change from a shutdown defense in depth approach to a shutdown probabilistic risk assessment approach).

(23) Maximum Extended Load Line Limit Analysis Plus (MELLLA+) Implementation

Prior to the first implementation of MELLLA+, TVA shall perform reload safety analyses using codes that have been corrected for the errors described in TVA letter CNL-19-125, dated December 19, 2019.

Insert 1

(24) TVA shall close all open Facts and Observations (F&Os) listed in Tables 11 and 13 to Attachment 2 of TVA Letter CNL-20-003, "Application for Technical Specifications Change Regarding Risk-Informed Justification for the Relocation of Specific Surveillance Frequency Requirements to a Licensee Controlled Program (BFN-TS-516)," dated March 27, 2020, prior to implementing any Surveillance Test Interval extensions under the Surveillance Frequency Control Program. The F&O closures will be performed in accordance with the ASME/ANS RA-Sa-2009 PRA Standard, as endorsed by Regulatory Guide 1.200.

D. The UFSAR supplement, as revised, submitted pursuant to 10 CFR 54.21(d), shall be included in the next scheduled update to the UFSAR required by 10 CFR 50.71(e)(4) following the issuance of this renewed operating license. Until that update is complete, TVA may make changes to the programs and activities described in the supplement without prior Commission approval, provided that TVA evaluates such changes pursuant to the criteria set forth in 10 CFR 50.59 and otherwise complies with the requirements in that section.

E. The UFSAR supplement, as revised, describes certain future activities to be complete prior to the period of extended operation. TVA shall complete these activities no later than December 20, 2013, and shall notify the NRC in writing when implementation of these activities is complete and can be verified by NRC inspection.

F. All capsules in the reactor vessel that are removed and tested must meet the test procedures and reporting requirements of the most recent NRC-approved version of the Boiling Water Reactor Vessels and Internals Project (BWRVIP) Integrated Surveillance Program (ISP) appropriate for the configuration of the specimens in the capsule. Any changes to the BWRVIP ISP capsule withdrawal schedule, including spare capsules, must be approved by the NRC prior to implementation. All capsules placed in storage

(23) Maximum Extended Load Line Limit Analysis Plus (MELLLA+) Implementation

Prior to the first implementation of MELLLA+, TVA shall perform reload safety analyses using codes that have been corrected for the errors described in TVA letter CNL-19-125, dated December 19, 2019.

Insert 1

(24) TVA shall close all open Facts and Observations (F&Os) listed in Tables 11 and 13 to Attachment 2 of TVA Letter CNL-20-003, "Application for Technical Specifications Change Regarding Risk-Informed Justification for the Relocation of Specific Surveillance Frequency Requirements to a Licensee Controlled Program (BFN-TS-516)," dated March 27, 2020, prior to implementing any Surveillance Test Interval extensions under the Surveillance Frequency Control Program. The F&O closures will be performed in accordance with the ASME/ANS RA-Sa-2009 PRA Standard, as endorsed by Regulatory Guide 1.200.

D. The UFSAR supplement, as revised, submitted pursuant to 10 CFR 54.21(d), shall be included in the next scheduled update to the UFSAR required by 10 CFR 50.71(e)(4) following the issuance of this renewed operating license. Until that update is complete, TVA may make changes to the programs and activities described in the supplement without prior Commission approval, provided that TVA evaluates such changes pursuant to the criteria set forth in 10 CFR 50.59 and otherwise complies with the requirements in that section.

E. The UFSAR supplement, as revised, describes certain future activities to be complete prior to the period of extended operation. TVA shall complete these activities no later than June 28, 2014, and shall notify the NRC in writing when implementation of these activities is complete and can be verified by NRC inspection.

F. All capsules in the reactor vessel that are removed and tested must meet the test procedures and reporting requirements of the most recent NRC-approved version of the Boiling Water Reactor Vessels and Internals Project (BWRVIP) Integrated Surveillance Program (ISP) appropriate for the configuration of the specimens in the

(16) Radiological Consequences Analyses Using Alternative Source Terms

TVA shall perform facility and licensing basis modifications to resolve the non-conforming/degraded condition associated with the Alternate Leakage Treatment pathway such that the current licensing basis dose calculations (approved in License Amendment Nos. 251/282 (Unit 1), 290/308 (Unit 2) and 249/267 (Unit 3)) would remain valid. These facility and licensing basis modifications shall be complete prior to initial power ascension above 3458.

(17) Prior to extending the frequency for the Integral Leakage Rate Testing described in TS 5.5.12, the licensee shall implement the modifications, that are modeled in the Fire PRA and described in Table S-2, Plant Modifications Committed, of Tennessee Valley Authority letter CNL-18-100, dated October 18, 2018; as supplemented by letter CNL-19-027, dated February 13, 2019.

(18) Maximum Extended Load Line Limit Analysis Plus (MELLLA+) Special Consideration

The licensee shall not operate the facility within the MELLLA+ operating domain more than a 10°F reduction in feedwater temperature below the design feedwater temperature.

(19) Maximum Extended Load Line Limit Analysis Plus (MELLLA+) Implementation

Prior to the first implementation of MELLLA+, TVA shall perform reload safety analyses using codes that have been corrected for the errors described in TVA letter CNL-19-125, dated December 19, 2019.

(20) TVA shall close all open Facts and Observations (F&Os) listed in Tables 11 and 13 to Attachment 2 of TVA Letter CNL-20-003, "Application for Technical Specifications Change Regarding Risk-Informed Justification for the Relocation of Specific Surveillance Frequency Requirements to a Licensee Controlled Program (BFN-TS-516)," dated March 27, 2020, prior to implementing any Surveillance Test Interval extensions under the Surveillance Frequency Control Program. The F&O closures will be performed in accordance with the ASME/ANS RA-Sa-2009 PRA Standard, as endorsed by Regulatory Guide 1.200.

Insert 1

D. The UFSAR supplement, as revised, submitted pursuant to 10 CFR 54.21(d), shall be included in the next scheduled update to the UFSAR required by 10 CFR 50.71(e)(4) following the issuance of this renewed operating license. Until that update is complete, TVA may make changes to the programs and activities described in the supplement without prior Commission approval, provided that TVA evaluates such changes pursuant to the criteria set forth in 10 CFR 50.59 and otherwise complies with the requirements in that section.

E. The UFSAR supplement, as revised, describes certain future activities to be complete prior to the period of extended operation. TVA shall complete these activities no later than July 2, 2016, and shall notify the NRC in writing when implementation of these activities is complete and can be verified by NRC inspection.