TVA TENNESSEE VALLEY AUTHORITY

1101 Market Street, Chattanooga, Tennessee 37402

CNL-22-084

September 16, 2022

10 CFR 50.90

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

Sequoyah Nuclear Plant, Units 1 and 2
Renewed Facility Operating License Nos. DPR-77 and DPR-79
NRC Docket Nos. 50-327 and 50-328

**Subject: Response to Request for Additional Information Regarding License Amendment Request to License Amendment Request to Modify Approved 10 CFR 50.69 Categorization Process (SQN-TS-21-07) (EPID L-2022-LLA-0033)**

References:

1. TVA letter to NRC, CNL-21-085, "License Amendment Request to Modify Approved 10 CFR 50.69 Categorization Process (SQN-TS-21-07)," dated February 24, 2022 (ML22055A625)

2. NRC electronic mail to TVA, "Request for Additional Information Related to SQN Request for Fire and Seismic PRA Modification to 50.69 L-2022-LLA-0033," dated August 2, 2022 (ML22214A158)

By letter dated February 24, 2022 (Reference 1), Tennessee Valley Authority (TVA) requested an amendment to the Sequoyah Nuclear Plant (SQN), Units 1 and 2 Renewed Facility Operating License (OL). The proposed amendment would modify the SQN OL to permit the use of the peer reviewed plant-specific SQN seismic probabilistic risk assessment and fire probabilistic risk assessment models into the previously approved Title 10 of the *Code of Federal Regulations* (10 CFR) 50.69 categorization process. In Reference 2, the Nuclear Regulatory Commission issued a request for additional information (RAI) and requested TVA respond by September 16, 2022.

The enclosure to this submittal provides the TVA response to the RAI.

TVA has reviewed the information supporting the no significant hazards consideration and the environmental consideration that was previously provided to the NRC in the referenced LAR. The additional information provided in this RAI response does not impact the conclusion that the proposed license amendment does not involve a significant hazards consideration. The additional information also does not impact the conclusion that there is no need for an environmental assessment to be prepared in support of the proposed amendment.

There are no new regulatory commitments contained in this submittal.

In accordance with 10 CFR 50.91, "Notice for Public Comment; State Consultation," a copy of this supplement is being provided to the Tennessee Department of Environment and Conservation.

Please address any questions regarding this submittal to Stuart L. Rymer, Senior Manager, Fleet Licensing, at slrymer@tva.gov.

I declare under penalty of perjury that the foregoing is true and correct. Executed on this 16th day of September 2022.

Respectfully,

Digitally signed by Edmondson, Carla
Date: 2022.09.16 11:46:45 -04'00'

James Barstow
Vice President, Nuclear Regulatory Affairs and Support Services

Enclosure: Response to Request for Additional Information

cc (w/ Enclosure):
- NRC Regional Administrator – Region II
- NRC Senior Resident Inspector – Sequoyah Nuclear Plant
- NRC Project Manager – Sequoyah Nuclear Plant
- Director, Division of Radiological Health – Tennessee State Department of Environment and Conservation

Enclosure

Response to Request for Additional Information

NRC Introduction

*By letter dated February 24, 2022 (ADAMS Accession No. ML22055A625), Tennessee Valley Authority (TVA, the licensee), submitted a license amendment request (LAR) regarding Sequoyah Nuclear Plant (SQN). The proposed license amendment would modify the SQN operation license to permit the use of the peer reviewed plant-specific seismic PRA and fire PRA models into the previously approved 10 CDFR 50.69 categorization process.*

Request for Additional Information

***RAI APLC-01 – Seismic Probabilistic Risk Assessment (SPRA) Upgrades during SPRA Fact and Observation (F&O) Closure Review***

*The American Society of Mechanical Engineers/American Nuclear Society (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," dated February 2009, 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 the ASME/ANS RA-Sa-2009 PRA Standard 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.*

*In Section 3.1.3 of the Enclosure to the LAR, the licensee stated that four seismic hazards analysis (SHA) findings were assessed to be upgrades and were resolved as part of a focused scope review. However, in the same section later, the licensee stated that "No upgrades were required as a part of the F&O closure." This sentence appears to contradict the previous statement. Although these findings were resolved, they were assessed to be upgrades as part of the F&O closure review process. It is noted that Section 5 of Enclosure 2 to the SQN TSTF-505 LAR (ADAMS Accession No. ML21217A174) contains a similar sentence "No additional upgrades were required as a part of the F&O closure review."*

*Explain the sentence "[n]o upgrades were required as a part of the F&O closure" in the February 24, 2022, LAR and confirm whether the sentence in TSTF-505 LAR would be more appropriate. If there were additional upgrades, please discuss these upgrades and the results of related focused-scope peer review(s).*

**TVA Response**

The original peer review of the SQN Units 1 and 2 SPRA was conducted at the TVA Chattanooga Office Complex (COC) in Chattanooga, Tennessee, during the week of April 23, 2018. The peer review report was issued in July 2018. Following the peer review, the SQN SPRA model and documentation were revised to resolve the Finding Level F&Os identified by the peer review team.

TVA identified that the revisions to the model and documentation made to resolve four of the F&Os did constitute upgrades as defined by the ASME/ANS RA-Sb-2013 PRA Standard. As a result, the SQN SPRA model required a Focused-Scope Peer Review (FSPR) and a F&O Closure Review to close out all open Finding Level F&Os. The SQN SPRA FSPR and F&O Independent Assessment were performed at the TVA COC, from February 4 - 8, 2019. The SQN

SPRA FSPR was performed in accordance with Nuclear Energy Institute (NEI) 05-04/12-13 "Seismic Probabilistic Risk Assessment (SPRA) Peer Review Process Guidelines." The F&O Independent Assessment was performed in accordance with Appendix X of NEI 05-04/12-13. The scope of the Independent Assessment included all the Finding Level F&Os from the prior PRA peer review and new findings generated in the FSPR.

Based on the upgrades performed on the SQN SPRA, the FSPR team determined that the scope of the review included supporting requirements (SRs) SHA-I2, SHA-J1, SHA-J2, and SHA-J3. SR SHA-I1 was assessed to be not applicable because this SR was judged not to be affected by the original four upgrade SHA findings. All four reviewed SRs were assessed to be met with three new F&Os. Subsequently, the SQN SPRA documentation was updated to address the three new F&Os.

The F&O Independent Assessment Team evaluated the resolutions from both the F&Os identified in the original peer review and the F&Os identified during the FSPR. Following the onsite review consensus sessions, a final consensus session on April 17, 2019 (via conference call) was conducted. In the final Independent Assessment report issued on April 22, 2019, all F&Os (including those generated in the FSPR) were designated as being closed under PRA maintenance. No additional upgrades were identified in the final assessment. That was the basis for the statement that no upgrades were required as a part of the F&O closure in both the February 24, 2022 LAR and the Technical Specifications Task Force (TSTF)-505 LAR.

***RAI APLC-02 – Integral Assessment***

*Paragraph (c)(1)(ii) of 10 CFR 50.69 requires that the SSC functional importance be determined using an integrated, systematic process. The categorization of SSCs, including those categorized using the SPRA, is based on importance measures and corresponding numerical criteria, as described in Section 5.1 of NEI 00-04. Section 5.6 of NEI 00-04, "Integral Assessment," discusses the need for an integrated computation using available importance measures. Section 5.6 further states that the "integrated importance measure essentially weights 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 Fussell-Vesely (FV) and integrated Risk Achievement Worth (RAW).*

*In Section 3.7.1 of the Enclosure to the LAR, the licensee stated that the importance evaluations performed in accordance with the process in NEI 00-04 are determined on a component basis and that most of the seismic and fire basic event importance measures can be directly aligned with components in the internal events (IE) PRA. The licensee provided the two exceptions: subcomponents and SSCs not in other PRA models.*

*Those seismic and/or fire basic events that are not explicitly modeled in the IE PRA, but function as subcomponents of components modeled in the IE PRA, will have their hazard specific importance measures combined with the other PRA importance measures. For those seismic and fire basic events that are not explicitly modeled in the IE PRA, an integrated safety significant computation is not necessary.*

a) *Describe how subcomponents modeled in the IE PRA are identified to link with a basic event in the SPRA or fire PRA (FPRA) and provide an example to illustrate the process.*

b) *Confirm that those seismic and fire basic events not explicitly modeled in the IE PRA will be directly reported to IDP if their importance measures are above the threshold based on NEI 00-04 guidance.*

**TVA Response**

**Response to Part a**

The importance evaluations performed in accordance with the process in NEI 00-04 are determined on a component basis. It is not necessary that there be complete alignment among the basic events that are pertinent to a given component from one hazard PRA to another, i.e., there may be hazard-specific basic events whose importance contributions are captured within the component importance calculations for that hazard.

A large majority of SPRA and FPRA basic events are directly aligned with the basic events in other PRA models and are combined using the formulae in Section 5.6 of NEI 00-04. However, as noted in this RAI, there are a few structures, systems, and components (SSC) in the SPRA and FPRA that are not directly included in the other PRA models.

Subcomponents

The importance of a subcomponent that was not directly modeled in other PRAs will be accounted for in the importance calculation for the component to which it is associated because it can be treated as another failure mode of that component. For example, the steam admission valve for the alternate feedwater pump is modeled in the Internal Events (IE) PRA, Internal Flooding (IF) PRA, FPRA, and the SPRA. Seismic-induced relay chatter, a failure mechanism unique to the SPRA, could cause the valve to close, stopping the flow of steam to the pump, which causes the pump to fail. This relay was considered part of the valve boundary within the IE and IF PRA models and its failures are inherently accounted for in the valve failure probability and associated component importance. For the SPRA, the relay was directly modeled to spuriously close the valve. The SPRA importance of the relay would be considered as a contributor to the valve failure and accounted for appropriately within the valve's importance measures for the integrated importance measures assessment, in accordance with the process in NEI 00-04.

Similarly, there is a FPRA basic event for the steam admission valve that accounts for fire damage on a cable associated with the steam admission valve. This allows the fire effects as well as random failures on the valve to be addressed. The fire-induced failure mechanisms would be considered as a contributor to the valve failure and accounted for appropriately within the valve’s importance measures for the integrated importance measures assessment, in accordance with NEI 00-04.

The decision to treat SPRA and FPRA basic events as representing subcomponents within the importance calculations for another modeled component will be made based on the modeling in each of the PRAs, as part of the PRA basic event-to-component mapping within the categorization process.

**Response to Part b**

While most of the SSCs in the SPRA and FPRA are directly aligned with SSCs in the IE and IF PRA models, there are some SSCs that are unique to the SPRA and/or the FPRA. These SSCs may have been screened out from the IE and IF models in accordance with the PRA modeling

requirements in the 2009 ASME/ANS PRA Standard, based on having no credible failure mode (or an extremely low probability of failure). If these SSCs are high safety significance (HSS) for the SPRA or the FPRA, then their integrated safety significance computation is not necessary. The safety significance would be presented to the Integrated Decision-making Panel (IDP) for consideration in the decision-making process. The NEI 00-04 process allows the IDP to adjust significance of a SPRA modeled SSC or a FPRA modeled SSC using proper justification. The quantitative integrated importance measure assessment is only one portion of the categorization process.

The following examples demonstrate how SSCs only in the SPRA would be treated for the importance analysis. Some components appear only in the SPRA, because they do not have a credible failure mode in other PRAs or have been screened out for other reasons.

- Structures are not directly included in the IE and IF PRA models because there is no credible failure mode for those hazards. However, some structures are included in the SPRA. If these structures are HSS in the SPRA, then their integrated safety significance computation is not necessary. The safety significance would be presented to the IDP for their consideration in the decision-making process. The NEI 00-04 §1.5 categorization process allows the IDP to adjust significance of a SPRA-modeled SSC using proper justification.

- SPRA specific systems and components: Some components appear only in the SPRA, because they do not have a credible failure mode in other PRA models or have been screened out for other reasons. These components will be treated as separate components for the integrated importance measure assessment. Examples are below.

    - Components such as cable trays, conduits, motor control centers, electrical cabinets and panels, HVAC (heating, ventilation, and air conditioning) ducting, and piping were not included in the IE PRA because they are passive components. However, for the SPRA, their seismic anchorage failure could potentially fail modeled components. For 50.69 categorization, the associated seismic failure events would be categorized based on their impact on modeled function. The integrated importance assessment would not change this categorization.

The following examples demonstrate how the components that are only in the FPRA would be treated for the importance analysis.

- Fire specific SSCs: Some components appear only in the FPRA, because they do not have a credible failure mode in other PRAs or have been screened out for other reasons. These components will be treated as separate components for the integral importance measure assessment. There are no structures or systems that are modeled only in the FPRA explicitly. The Fire protection detection and suppression systems are modeled implicitly within the calculation of the initiating event modeling, but they do not have a calculated importance measure for each component credited, instead relying on the initiator to view how important those components are.

    - Main Control Room abandonment is not considered in the IE, IF, or SPRA models. This is a unique failure mode that is accounted for in the FPRA to account for Main Control Room abandonment due to habitability or other fire concerns. Each of the components required following abandonment may or may not be included in the other PRA models. If the components are modeled with other hazards, the impact

will be included in the importance measure assessments as described above. If they are not included, then the component will be mapped to the appropriate component and categorized accordingly.

- Components such as the excess letdown isolation valves appear only in the FPRA due to the component being screened out from the IE, IF, and SPRA models due to not having credible failure modes (multiple valves would need to fail and the random failure of each valve is sufficiently low that they would be screened out from the analysis) for those hazards.

In summary, most of the seismic and fire hazards basic event importance measures can be directly aligned with components in the other PRA models. Those seismic and fire basic events that are not explicitly modeled in other PRA models, but function as subcomponents of components modeled in other PRA models, will have their seismic importance measures combined with the other model's importance measures using the NEI 00-04 §5.6 formulae for the integral assessment. An integrated safety significance computation is not necessary for other seismic or fire basic events that are not explicitly modeled in the IE or IF PRAs because the integrated significance computation is only performed if an SSC modeled in the SPRA or FPRA has an initial HSS ranking. The safety significance would be presented to the IDP for consideration in the decision-making process. The NEI 00-04 categorization process allows the IDP to adjust significance of an SPRA-modeled SSC using proper justification.

***RAI APLC-03 – PRA Model Uncertainty Dispositions***

*Paragraphs 50.69(c)(1)(i) and (c)(1)(ii) of 10 CFR require that a licensee's PRA be of sufficient quality and level of detail to support the 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. The guidance in NEI 00-04 specifies sensitivity studies to be conducted for each PRA model to address uncertainty. The sensitivity studies are performed to ensure that assumptions and sources of uncertainty (e.g., human error, common-cause failure, and maintenance unavailability) do not mask importance of components. The guidance in NEI 00-04 states that additional "applicable sensitivity studies" from characterization of PRA adequacy should be considered.*

*In Section 3.6 of the Enclosure to the LAR, the licensee stated that none of the PRA uncertainties or assumptions from Appendix A were identified as potential key assumptions and sources of uncertainty for the 10 CFR 50.69 application. It is unclear to the NRC staff which Appendix A is being referred to because the LAR does not include an Appendix A.*

*Provide the list of the seismic PRA assumptions and uncertainties and their dispositions with the conclusion if there are key assumptions and sources of uncertainty for the 10 CFR 50.69 application. For any key assumptions and sources of uncertainty for this application, discuss how they will be considered in categorization consistent with the guidance in NEI 00-04.*

[Note – This RAI was later clarified with NRC staff that a listing of the SPRA non-key assumptions and uncertainties with individual dispositions was not needed for this response.]

**TVA Response**

The referenced Appendix A in Section 3.6 was referring to Report SQN-0-21-126 developed for TVA by a contractor. Appendix A "Review of PRA Assumptions" was performed on all the PRA assumptions in the internal events, fire and seismic models. Each of these assumptions were then screened based on the screening criteria specified in Section 3.6 of the Enclosure to the LAR (CNL-21-085). These criteria are also specified below. If any of the assumptions or uncertainties were not able to be screened out, that assumption or uncertainty was considered key with respect to the 10 CFR 50.69 application. However, based on the review performed, none of the assumptions or uncertainties in Appendix A were identified as being key with respect to the 10 CFR 50.69 application.

**Criterion 1** - The uncertainty is addressed by implementing a "consensus model" defined as follows:

> Consensus model – In the most general sense, a consensus model is a model that has a publicly available published basis and has been peer reviewed and widely adopted by an appropriate stakeholder group. In addition, widely accepted PRA practices may be regarded as consensus models. Examples of the latter include the use of the constant probability of failure on demand model for standby components and the Poisson model for initiating events. For risk-informed regulatory decisions, the consensus model approach is one that NRC has utilized or accepted for the specific risk-informed application for which it is proposed. (Reference - NUREG-1855).
>
> Consensus method/model – In the context of risk-informed regulatory decisions, a method or model approach that the NRC has used or accepted for the specific risk-informed application for which it is proposed. A consensus method or model may also have a publicly available, published basis and may have been peer reviewed and widely adopted by an appropriate stakeholder group. (Reference - Regulatory Guide (RG) 1.200).
>
> Electric Power Research Institute (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 for which there is extensive historical precedent, and which produces results that are reasonable and realistic, can be screened out from further consideration. According to NRC Regulatory Position C.3.3.2 in RG 1.200, "When a key assumption is shown to be consistent with a consensus method or approach, that key assumption may no longer be subject to additional sensitivity studies in the context of a PRA application."

**Criterion 2** - The uncertainty has no impact or insignificant impact on the PRA results and therefore no impact or insignificant impact on the calculated change in risk due to proposed changes that are to be addressed by the PRA application.

**Criterion 3** - The assumption 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 out from further consideration.

**Criterion 4 -** There is no reasonable alternative assumption or reasonable modeling refinement to address the uncertainty that would produce different results. For the base PRA, the term “different results” refers to a change in the risk profile (e.g., total core damage frequency (CDF) and total large early release frequency (LERF), or the set of initiating events and accident sequences that contribute most to CDF and to LERF) and the associated changes in insights derived from the changes in the risk profile. A “reasonable alternative” assumption is one that has broad acceptance within the technical community and for which the technical basis for consideration is at least as sound as that of the assumption being challenged. (References - NUREG-1855, ASME/ANS RA-Sa-2009).

**Criterion 5 -** There is no reasonable alternative assumption or reasonable modeling refinement to address the uncertainty that is at least as sound as the assumption under consideration. A “reasonable alternative” assumption is one that has broad acceptance within the technical community and for which the technical basis for consideration is at least as sound as that of the assumption being challenged. (References - NUREG-1855, ASME/ANS RA-Sa-2009).