UNITED STATES
NUCLEAR REGULATORY COMMISSION
WASHINGTON, D.C. 20555-0001

July 17, 2020

Mr. James Barstow
Vice President, Nuclear Regulatory Affairs
and Support Services
Tennessee Valley Authority
Sequoyah Nuclear Plant
1101 Market Street, LP 4A-C
Chattanooga, TN 37402-2801

SUBJECT: SEQUOYAH NUCLEAR PLANT, UNITS 1 AND 2 – STAFF REVIEW OF SEISMIC PROBABILISTIC RISK ASSESSMENT ASSOCIATED WITH REEVALUATED SEISMIC HAZARD IMPLEMENTATION OF THE NEAR-TERM TASK FORCE RECOMMENDATION 2.1: SEISMIC (EPID NO. L-2019-JLD-0013)

Dear Mr. Barstow:

The purpose of this letter is to document the staff's evaluation of the Sequoyah Nuclear Plant, Units 1 And 2 (Sequoyah), seismic probabilistic risk assessment (SPRA) which was submitted in response to Near-Term Task Force (NTTF) Recommendation 2.1 "Seismic." The U.S. Nuclear Regulatory Commission (NRC) has concluded that no further response or regulatory actions associated with NTTF Recommendation 2.1 "Seismic" are required for Sequoyah.

By letter dated March 12, 2012 (Agencywide Documents Access and Management System (ADAMS) Accession No. ML12053A340), the NRC issued a request for information under Title 10 of the *Code of Federal Regulations* Section 50.54(f) (hereafter referred to as the 50.54(f) letter). The request was issued as part of implementing lessons learned from the accident at the Fukushima Dai-ichi nuclear power plant. Enclosure 1 to the 50.54(f) letter requested that licensees reevaluate seismic hazards at their sites using present-day methodologies and guidance. Enclosure 1, Item (8), of the 50.54(f) letter requested that certain licensees complete an SPRA to determine if plant enhancements are warranted due to the change in the reevaluated seismic hazard compared to the site's design-basis seismic hazard.

By letter dated October 18, 2019 (ADAMS Accession No. ML19291A003), Tennessee Valley Authority (TVA, the licensee), provided its SPRA submittal in response to Enclosure 1, Item (8) of the 50.54(f) letter for Sequoyah. The NRC staff assessed the licensee's implementation of the Electric Power Research Institute's Report 1025287, "Seismic Evaluation Guidance - Screening, Prioritization, and Implementation Details (SPID) for the Resolution of Fukushima Near-Term Task Force Recommendation 2.1: Seismic" (ADAMS Accession No. ML12333A170), as endorsed by NRC letter dated February 15, 2013 (ADAMS Accession No. ML12319A074), through the completion of the reviewer checklist in Enclosure 1 to this letter. As described below, the NRC staff has concluded that the Sequoyah SPRA submittal meets the intent of the SPID guidance and that the results and risk insights provided by the SPRA support the NRC's determination that no further response or regulatory actions associated with NTTF Recommendation 2.1 "Seismic" are required.

BACKGROUND

The 50.54(f) letter requested, in part, that licensees reevaluate the seismic hazards at their sites using updated hazard information and current regulatory guidance and methodologies. The request for information and the subsequent NRC evaluations have been divided into two phases:

> **Phase 1**: Issue 50.54(f) letters to all operating power reactor licensees to request that they reevaluate the seismic and flooding hazards at their sites using updated seismic and flood hazard information and present-day regulatory guidance and methodologies and, if necessary, to request they perform a risk evaluation.
>
> **Phase 2**: Based upon the results of Phase 1, the NRC staff will determine whether additional regulatory actions are necessary (e.g., updating the design basis and structures, systems, and components important to safety) to provide additional protection against the updated hazards.

By letter dated March 31, 2014 (ADAMS Accession No. ML14098A478), TVA submitted the reevaluated seismic hazard information for Sequoyah. The NRC staff conducted an assessment of the submittal and issued a response letter on April 27, 2015 (ADAMS Accession No. ML15098A641). The NRC staff's assessment concluded that TVA conducted the hazard reevaluation using present-day regulatory guidance and methodologies, appropriately characterized the site, and met the intent of the guidance for determining the reevaluated seismic hazard at Sequoyah.

By letter dated October 27, 2015 (ADAMS Accession No. ML15194A015), the NRC documented a determination of which licensees were to perform: (1) an SPRA; (2) limited scope evaluations; or (3) no further actions, based on, among other factors, a comparison of the reevaluated seismic hazard and the site's design-basis earthquake. As documented in that letter, Sequoyah was expected to complete an SPRA with an estimated completion date of December 31, 2019, which would also assess high frequency ground motion effects. In addition, TVA was expected to perform a limited-scope evaluation for the spent fuel pool (SFP), which was submitted by letter dated December 23, 2016 (ADAMS Accession No. ML16362A204). The staff provided its assessment of the Sequoyah SFP evaluation by letter dated February 16, 2017 (ADAMS Accession No. ML17041A387).

The completion of the April 27, 2015, NRC staff assessment for the reevaluated seismic hazard and the scheduling of the Sequoyah SPRA submittal as described in the NRC's letter dated October 27, 2015, marked the fulfillment of the Phase 1 process for Sequoyah.

In its letter dated October 18, 2019, TVA provided the SPRA submittal that initiated the NRC's Phase 2 decisionmaking process for Sequoyah. The NRC described this Phase 2 decisionmaking process in a guidance memorandum from the Director of the Division of Operating Reactor Licensing to the Director of the Office of Nuclear Reactor Regulation (NRR) dated March 2, 2020 (ADAMS Accession No. ML20043D958). This memorandum describes a Senior Management Review Panel (SMRP) consisting of three NRR Division Directors that are expected to reach a screening decision for each plant submitting an SPRA. The SMRP is supported by appropriate technical staff who are responsible for consolidating relevant information and developing screening recommendations for consideration by the panel. In presenting recommendations to the SMRP, the supporting technical staff is expected to recommend placement of each SPRA plant into one of three groups:

1) **Group 1** includes plants for which available information indicates that further regulatory action is not warranted. For seismic hazards, Group 1 includes plants for which the mean seismic core damage frequency (SCDF) and mean seismic large early release frequency (SLERF) clearly demonstrate that a plant-specific backfit would not be warranted.

2) **Group 2** includes plants for which further regulatory action should be considered under the NRC's backfit provisions. This group may include plants with relatively large SCDF or SLERF, such that the event frequency in combination with other factors results in a risk to public health and safety for which a regulatory action is expected to provide a substantial safety enhancement.

3) **Group 3** includes plants for which further regulatory action may be needed, but more thorough consideration of both qualitative and quantitative risk insights is needed before determining whether a formal backfit analysis is warranted.

The evaluation performed to provide the basis for the staff's grouping recommendation to the SMRP for Sequoyah is described below. Based on its evaluation, the staff recommended to the SMRP that Sequoyah be classified as a Group 1 plant and therefore, no further regulatory action was warranted.

EVALUATION

Upon receipt of the licensee's October 18, 2019, SPRA report, a technical team of NRC staff members performed a completeness review to determine if the necessary information to support Phase 2 decisionmaking had been included in the licensee's submittal. The technical team performing the review consisted of staff experts in the fields of seismic hazards, fragilities evaluations, and plant response/risk analysis. On November 19, 2019 (ADAMS Accession No. ML19323E793), the technical team determined that sufficient information was available to perform the detailed technical review in support of the Phase 2 decision.

As described in the 50.54(f) letter, the staff's detailed review focused on verifying the technical adequacy of the licensee's SPRA such that an appropriate level of confidence could be placed in the results and risk insights of the SPRA to support regulatory decisionmaking associated with the 50.54(f) letter. As stated in its October 18, 2019, submittal, the licensee developed and documented the SPRA in accordance with the SPID guidance, including performing a full-scope peer review against Part 5 of Addendum B to the American Society of Mechanical Engineers (ASME)/American Nuclear Society (ANS), "Standard for Level 1/Large Early Release Frequency Probabilistic Risk Assessment for Nuclear Power Plant Applications," (RA-Sb-2013). In addition, the licensee also performed a close-out independent assessment of the resolution of the finding-level facts and observations (F&Os) from the full-scope peer review following the process accepted by the NRC (ADAMS Accession No. ML17079A427). The close-out independent assessment also included a concurrent focused-scope peer review for upgrades to the SPRA. The close-out independent assessment resulted in the closure of all finding-level F&Os for the Sequoyah SPRA. Appendix A of the licensee's submittal provided a summary of the full-scope and independent assessment peer reviews, including excerpts from the corresponding peer review reports.

By letter dated July 6, 2017 (ADAMS Accession No. ML17177A446), the NRC issued a generic audit plan and entered into the audit process described in Office Instruction LIC -111, “Regulatory Audits,” dated December 29, 2008 (ADAMS Accession No. ML082900195), to assist in the timely and efficient closure of activities associated with the 50.54(f) letter. The list of applicable licensees in Enclosure 1 of the July 6, 2017, letter included TVA as the licensee for Sequoyah. The NRC staff exercised the audit process by reviewing selected licensee documents via an electronic reading room (eportal) as documented in Enclosure 3 to this letter.

The full-scope peer review and the close-out independent assessment reports were available to the NRC staff on the eportal. The staff sampled the resolutions to the findings as well as the close-out independent assessment team’s conclusions from the close-out report.

Since the licensee’s internal events PRA (IEPRA) model was used as the basis for the development of the SPRA model, the NRC staff also reviewed the IEPRA F&Os and the associated dispositions during the SPRA audit process to assess any potential impact on the SPRA submittal. The technical adequacy of the IEPRA is described in Appendix A of the SPRA submittal, including the F&O closure review results showing that all Finding-level F&Os from the full-scope IEPRA peer review have been closed. The staff also previously found the IEPRA to be technically acceptable for the licensee to implement 10 CFR 50.69 as described in the NRC staff safety evaluation dated September 18, 2019 (ADAMS Accession No. ML19179A135). The NRC staff reviewed the IEPRA F&O dispositions and did not identify any modeling issues that could impact the conclusions of the SPRA submittal.

During the audit process, the staff developed questions to clarify information in the licensee’s submittal and to gain understanding of non-docketed information that supports the docketed SPRA submittal. The staff’s clarification questions dated November 19, 2019, February 19, 2020, and April 9, 2020 (ADAMS Accession Nos. ML19323E793, ML20056D696, and ML20161A390, respectively), were sent to the licensee to support the audit. The licensee subsequently provided answers to the questions via the eportal, which the staff reviewed.

The staff determined that the answers to the audit questions provided in the eportal served to confirm statements that the licensee made in its October 18, 2019, SPRA submittal. Based on the staff’s review of the licensee’s submittal, including the resolution of the peer review findings as described above, the NRC staff concluded that the technical adequacy of the licensee’s SPRA submittal was sufficient to support regulatory decisionmaking associated with Phase 2 of the 50.54(f) letter.

The staff’s review process included the completion of the SPRA Submittal Technical Review Checklist (SPRA Checklist) contained in Enclosure 1 to this letter. As described in Enclosure 1, the SPRA Checklist is a document used to record the staff’s review of licensees’ SPRA submittals against the applicable guidance of the SPID in response to the 50.54(f) letter. The SPRA Checklist also focuses on areas where the SPID contains differing guidance from standard industry SPRA guidance. Enclosure 1 contains the staff’s application of the SPRA checklist to Sequoyah’s submittal. As documented in the checklist, the staff concluded that the Sequoyah SPRA met the intent of the SPID. The staff further concluded that the peer review was done in accordance with the ASME/ANS Standard RA-Sb-2013 process and the peer review findings have been closed-out using the NRC-accepted independent assessment process outlined in Appendix X to Nuclear Energy Institute (NEI) guidance document NEI 12-13.

Following the staff’s conclusion on the SPRA’s technical adequacy, the staff reviewed the risk and safety insights contained in the Sequoyah SPRA submittal. The staff used the screening criteria described in a staff memorandum dated August 29, 2017 (ADAMS Accession No. ML17146A200), titled, "Guidance for Determination of Appropriate Regulatory Action Based on Seismic Probabilistic Risk Assessment Submittals in Response to Near Term Task Force Recommendation 2.1: Seismic" to guide its review and screening recommendation to the SMRP. The criteria in the staff's guidance document includes thresholds to assist in determining whether to apply the backfit screening process described in Management Directive 8.4, "Management of Facility Specific Backfitting, Forward Fitting, Issue Finality, and Information Requests," dated September 20, 2019 (ADAMS Accession No. ML18093B087), to the SPRA submittal review. As part of this review, the staff considered potential modifications that could help identify substantial safety enhancements that could be cost-justified. The NRC staff used the SCDF and SLERF results provided in the Sequoyah SPRA submittal and other available information in conjunction with the guidance in the staff memorandum dated August 29, 2017, to complete a detailed screening evaluation. The detailed screening concluded that Sequoyah should be considered a Group 1 plant because:

- The staff did not identify any potential modifications that would be considered necessary for adequate protection or compliance with existing requirements;

- Sufficient reductions in SCDF and SLERF cannot be achieved by potential modifications considered in this evaluation to constitute cost-justified substantial safety improvements based upon importance measures, available information, and engineering judgement; and

- Additional consideration of containment performance, as described in NUREG/BR-0058, "Regulatory Analysis Guidelines of the U.S. Nuclear Regulatory Commission," does not identify a modification that would result in a substantial safety improvement.

A discussion of the detailed screening evaluation completed by the NRC staff is provided in Enclosure 2 to this letter.

Based on the detailed screening evaluation and its review of the Sequoyah SPRA submittal, the technical team determined that recommending Sequoyah to be classified as a Group 1 plant was appropriate and additional review and/or analysis to pursue a plant-specific backfit was not warranted.

As a part of the Phase 2 decisionmaking process for SPRAs, the NRC formed the Technical Review Board (TRB), a board of senior-level NRC subject matter experts, to ensure consistency of review across the spectrum of plants that will be providing SPRA submittals. The technical review team provided the results of the Sequoyah review to the TRB with the Phase 2 recommendation that Sequoyah be categorized as a Group 1 plant, meaning that no further response or regulatory actions are required. The TRB members assessed the information presented by the technical team and agreed with the team's recommendation for classification of Sequoyah as a Group 1 plant.

Subsequently, the technical review team met with the SMRP and presented the results of the review including the recommendation for Sequoyah to be categorized as a Group 1 plant. The SMRP members asked questions about the review, as well as the risk insights and provided input to the technical review team. The SMRP approved the staff's recommendation that Sequoyah should be classified as a Group 1 plant, meaning that no further response or regulatory action is required.

AUDIT REPORT

The generic audit plan dated July 6, 2017, describes the NRC staff's intention to issue an audit report that summarizes and documents the NRC's regulatory audit of a licensee's SPRA submittal associated with their reevaluated seismic hazard information. The NRC staff's audit included a review of licensee documents through an electronic reading room. An audit summary document is included as Enclosure 3 to this letter.

CONCLUSION

Based on the staff's review of the Sequoyah submittal against the endorsed SPID guidance, the NRC staff concludes that the licensee responded appropriately to Enclosure 1, Item (8) of the 50.54(f) letter. Additionally, the staff's review concluded that the SPRA is of sufficient technical adequacy to support Phase 2 regulatory decisionmaking in accordance with the intent of the 50.54(f) letter. Based on the results and risk insights of the SPRA submittal, the NRC staff also concludes that no further response or regulatory actions associated with NTTF Recommendation 2.1 "Seismic" are required.

Application of this review is limited to the review of the 10 CFR 50.54(f) response associated with NTTF Recommendation 2.1 "Seismic" review. The staff notes that assessment of the SPRA for use in other licensing applications, would warrant review of the SPRA for its intended application. The NRC may use insights from this SPRA assessment in its regulatory activities as appropriate.

If you have any questions, please contact Stephen Philpott at (301) 415-2365 or via e-mail at Stephen.Philpott@nrc.gov.

Sincerely,

/RA/

Mohamed K. Shams, Deputy Director
Division of Operating Reactor Licensing
Office of Nuclear Reactor Regulation

Docket Nos. 50-327 and 50-328

Enclosures:
1. NRC Staff SPRA Submittal Technical Review Checklist
2. NRC Staff SPRA Submittal Detailed Screening Evaluation
3. NRC Staff Audit Summary

cc w/encls: Listserv

NRC Staff SPRA Submittal Technical Review Checklist

Several nuclear power plant licensees have performed seismic probabilistic risk assessments (SPRAs) as part of their required submittals to satisfy Near-Term Task Force (NTTF) Recommendation 2.1: Seismic. These submittals were prepared according to the guidance in the Electric Power Research Institute – Nuclear Energy Institute (EPRI-NEI) Screening, Prioritization, and Implementation Details (SPID) document (EPRI-SPID, 2012), which was endorsed by the U.S. Nuclear Regulatory Commission (NRC) staff for this purpose. The SPRA peer reviews are also expected to follow the guidance in NEI 12-13 (NEI, 2012).

The SPID indicates that an SPRA submitted for the purpose of satisfying NTTF Recommendation 2.1: Seismic (hereafter referred to as NTTF Recommendation 2.1) must meet the requirements in the American Society of Mechanical Engineers-American Nuclear Society (ASME-ANS) PRA Methodology Standard (the ASME-ANS Standard). Either the "Addendum A version" (ASME/ANS Addendum A, 2009) or the "Addendum B version" (ASME/ANS Addendum B, 2013) of the ASME-ANS Standard can be used.

Tables 6-4, 6-5, and 6-6 of the SPID also provide a comparison of each of the Supporting Requirements (SRs) of the ASME/ANS Standard to the relevant guidance in the SPID. For most SRs, the SPID guidance does not differ from the requirement in the ASME/ANS Standard. However, because the guidance of the SPID and the criteria of the ASME/ANS Standard differ in some areas, or the SPID does not explicitly address an SR, the staff developed this checklist, in part, to help staff members to address and evaluate the differences.

In general, the SPID allowed departures or differed from the ASME/ANS Standard in the following ways:

(i) In some technical areas, the SPID's requirements tell the SPRA analyst "how to perform" one aspect of the SPRA analysis, whereas the ASME/ANS Standard's requirements generally cover "what to do" rather than "how to do it."

(ii) For some technical areas and issues, the requirements in the SPID differ from those in the ASME/ANS Standard.

(iii) The SPID has some requirements that are not in the ASME/ANS Standard.

The technical positions in the SPID have been endorsed by the NRC staff for NTTF Recommendation 2.1 submittals, subject to certain conditions concerning peer review outlined in the staff's November 12, 2012, letter to NEI (NRC, 2012).

The following checklist is comprised of the 16 "Topics" that require additional staff guidance because the SPID contains specific guidance that differs from the ASME/ANS Standard or expands on it. Each is covered below under its own heading, "Topic 1," "2," etc. This checklist was discussed during a public meeting held on December 7, 2016 (ADAMS Accession No. ML16350A181).

- Topic 1: Seismic Hazard (SPID Sections 2.1, 2.2, and 2.3)
- Topic 2: Site Seismic Response (SPID Section 2.4)

- Topic 3: Definition of the Control Point for the Safe Shutdown Earthquake (SSE) to Ground Motion Response Spectrum (GMRS) Comparison Aspect of the Site Analysis (SPID Section 2.4.2)
- Topic 4: Adequacy of the Structural Model (SPID Section 6.3.1)
- Topic 5: Use of Fixed-Based Dynamic Seismic Analysis of Structures for Sites Previously Defined as “Rock” (SPID Section 6.3.3)
- Topic 6: Use of Seismic Response Scaling (SPID Section 6.3.2)
- Topic 7: Use of New Response Analysis for Building Response, In-Structure Response Spectra (ISRS), and Fragilities
- Topic 8: Screening by Capacity to Select Structures, Systems, and Components (SSCs) for Seismic Fragility Analysis (SPID Section 6.4.3)
- Topic 9: Use of the Conservative Deterministic Failure Margin (CDFM)/Hybrid Methodology for Fragility Analysis (SPID Section 6.4.1)
- Topic 10: Capacities of SSCs Sensitive to High-Frequencies (SPID Section 6.4.2)
- Topic 11: Capacities of Relays Sensitive to High-Frequencies (SPID Section 6.4.2)
- Topic 12: Selection of Dominant Risk Contributors that Require Fragility Analysis Using the Separation of Variables (SOV) Methodology (SPID Section 6.4.1)
- Topic 13: Evaluation of Large Early Release Frequency (LERF) (SPID Section 6.5.1)
- Topic 14: Peer Review of the SPRA, Accounting for NEI 12-13 (SPID Section 6.7)
- Topic 15: Documentation of the SPRA (SPID Section 6.8)
- Topic 16: Review of Plant Modifications and Licensee Actions

TOPIC 1: Seismic Hazard (SPID Sections 2.1, 2.2, and 2.3)

The site under review has updated/revised its Probabilistic Seismic Hazard Analysis (PSHA) from what was submitted to NRC in response to the NTTF Recommendation 2.1: Seismic 50.54(f) letter.	No
Notes from staff reviewer: None. Deviation(s) or deficiency(ies) and Resolution: None. Consequence(s): N/A	
The NRC staff concludes that:	
• The peer review findings have been addressed and the analysis approach has been accepted by the staff for the purposes of this evaluation. The peer review findings referred to relate to the Probabilistic Seismic Hazards Analysis (SHA) requirements in the ASME/ANS Standard, as well as to the requirements in the SPID.	Yes
• Although some peer review findings have not been resolved, the analysis is acceptable on another justified basis.	N/A
• The guidance in the SPID was followed for developing the probabilistic seismic hazard for the site.	Yes
• An alternate approach was used and is acceptable on a justified basis.	N/A

TOPIC 2: Site Seismic Response (SPID Section 2.4)

<table>
<tr><td>The site under review has updated/revised its site response analysis from what was submitted to NRC in response to the NTTF Recommendation 2.1: Seismic 50.54(f) letter.</td><td>Yes</td></tr>
<tr><td>The guidance in the SPID was followed for developing a site profile for use in the analysis to develop control point seismic hazard curves (site response).</td><td>Yes</td></tr>
<tr><td colspan="2">Notes from staff reviewer: Tennessee Valley Authority (TVA, the licensee) updated its site velocity profiles to more accurately represent the subsurface. These velocity changes had a minimal impact on the site-specific hazard. The licensee performed an evaluation for liquefaction and lateral spreading for several SSCs. Peer review findings for items related to the liquefaction analysis have been closed through the NEI 12-13 Appendix X process.

Deviation(s) or deficiency(ies) and Resolution: None.

Consequence(s): N/A</td></tr>
<tr><td>The NRC staff concludes that:</td><td></td></tr>
<tr><td>• The peer review findings have been addressed and the analysis approach has been accepted by the staff for the purposes of this evaluation. The peer review findings referred to relate to the SRs SHA-E1 and E2 in the ASME/ANS Standard, as well as to the requirements in the SPID.</td><td>Yes</td></tr>
<tr><td>• Although some peer review findings have not been resolved, the analysis is acceptable on another justified basis.</td><td>N/A</td></tr>
<tr><td>• The licensee’s development of PSHA inputs and base rock hazard curves meets the intent of the SPID guidance or another acceptable approach.</td><td>Yes</td></tr>
<tr><td>• The licensee’s development of a site profile for use in the analysis adequately meets the intent of the SPID guidance or another acceptable approach.</td><td>Yes</td></tr>
<tr><td>• Although the licensee’s development of a V_s velocity profile for use in the analysis does not meet the intent of the SPID guidance, it is acceptable on another justified basis.</td><td>N/A</td></tr>
</table>

TOPIC 3: Definition of the Control Point for the SSE to GMRS Comparison Aspect of the Site Analysis (SPID Section 2.4.2)

The issue is establishing the control point where the safe shutdown earthquake (SSE) is defined. Most sites have only one SSE, but some sites have more than one SSE, for example one at rock and one at the top of the soil layer. This control point is needed because it is used as part of the input information for the development of the seismic site-response analysis, which in turn is an important input for analyzing seismic fragilities in the SPRA. The SPID (Section 2.4.1) recommends one of two criteria for establishing the control point for a logical SSE-to-GMRS comparison:	
A) If the SSE control point(s) is defined in the final safety analysis report (FSAR), it should be used as defined.	Yes
B) If the SSE control point is not defined in the FSAR, one of three criteria in the SPID (Section 2.4.1) should be used.	N/A
C) An alternative method has been used for this site.	N/A
The control point used as input for the SPRA is identical to the control point used to establish the GMRS.	Yes
If yes, the control point can be used in the SPRA and the NRC staff's earlier acceptance governs.	
If no, the NRC staff's previous reviews might not apply. The staff's review of the control point used in the SPRA is acceptable.	N/A
Notes from staff reviewer: None. Deviation(s) or deficiency(ies) and Resolution: None. Consequence(s): N/A	

The NRC staff concludes that:	
• The peer review findings have been addressed and the analysis approach has been accepted by the staff for the purposes of this evaluation. The peer review findings referred to relate to the requirements in the SPID. No requirements in the ASME/ANS Standard specifically address this topic.	N/A
• Although some peer review findings have not been resolved, the analysis is acceptable on another justified basis.	N/A
• The licensee's definition of the control point for site response analysis adequately meets the intent of the SPID guidance.	Yes
• The licensee's definition of the control point for site response analysis does not meet the intent of the SPID guidance, but is acceptable on another justified basis.	N/A

TOPIC 4: Adequacy of the Structural Model (SPID Section 6.3.1)

The NRC staff review of the structural model finds an acceptable demonstration of its adequacy.	Yes
Used an existing structural model	Yes
Used an enhancement of an existing model	No
Used an entirely new model	Yes
Criteria 1 through 7 (SPID Section 6.3.1) are all met.	Yes

Notes from staff reviewer:

Section 4.3 of the Sequoyah Nuclear Plant, Units 1 and 2 (Sequoyah) SPRA report describes the analysis of structures which support the safety-related components and systems. Table 4.3-1 of the report provides a summary of the foundation condition, structural modeling, and the structural analytical approaches used for the Auxiliary-Control Building (ACB), Reactor Building (RB), Essential Raw Cooling Water Pumping Station (ERCW), Additional Equipment Buildings (AEBs), East Steam Valve Room (ESVR), Diesel Generator Building (DGB), and Additional Diesel Generator Building (ADGB). New finite element models were developed for the ACB, ERCW, AEBs, ESVR, and DGB, while lumped-mass-stick model (LMSM) was used for RB and ADGB. The Sequoyah submittal states that the existing LMSM was used for the ADGB and explained that the RB LMSM model includes the Internal Concrete Structure, Concrete Shield Building, Steel Containment Vessel, and nuclear steam supply system (NSSS) components. A combined 3D finite element model (FEM) of the ESVR and the LMSM of the RB mounted on 3D FEM foundation slab was analyzed for seismic response.

The SPRA report explains that Sequoyah is a firm rock site. The ACB, RB, ERCW, AEBs are founded on rock; however, the DGB and ADGB are founded on soil, and the slab foundation of the ESVR is supported by caissons socketed to the bedrock. The licensee performed probabilistic soil structure interaction analyses for all buildings for evaluating seismic response and in-structure response spectra. The results from the probabilistic structural analysis were used to develop 50th percentile and 84th percentile in-structure response spectra at the locations of the seismic equipment list (SEL) systems and components.

In response to one of the finding-level facts and observations (F&O) from the full-scope peer review, F&O 7-13, the licensee addressed closure of gaps and potential pounding between the RB and the auxiliary building. The licensee's evaluation included calculation of displacement of the buildings and fragility of relay groups impacted by the pounding between the buildings. The F&Os were closed in accordance with the NEI 12-13 Appendix X independent assessment process.

The SPRA report states that configuration of some of the major structures of Sequoyah and Watts Bar Nuclear Plant (Watts Bar) are similar and thus the structural models of Watts Bar were used for the Sequoyah structural response analysis. A staff audit of

Sequoyah documents identified that the Watts Bar structural models were modified to include the local geometric differences.

The Sequoyah submittal discussed soil failure and fragility analysis in Section 3.3. Table 3.3-1 therein listed the structures and the underlaying foundation geotechnical materials. The structures founded on rock were screened out from potential failure of foundation material. For those SSCs founded on soil, the licensee evaluated the effect of soil failure modes considering liquefaction, seismic induced settlements, seismic induced lateral deformation, slope stability, sliding of earth and building structures, and seismic bearing capacity. The licensee determined that ERCW piping was susceptible to differential soil movement and developed fragility parameters using the separation of variables method discussed in Topic 12 of this checklist.

Section 4.3.3.1 of the submittal addresses Criteria 1 through 7 (SPID Section 6.3.1) for LMSMs of the RB and ADGB. The NRC staff evaluated the structural modeling and response analysis as part of the audit and confirmed that the LMSMs capture structural responses, torsional effects resulting from eccentricities, and in-plane floor flexibility. The cut-off frequency ranged from 20 to 50 hertz (Hz). Based on the audit results, the NRC staff concluded that appropriate modes of vibration of the structures were considered in the analysis and that the modeling approach applied the requirements of ASCE-4. Thus, the NRC staff finds that SPID Section 6.3.1 criterial 1 through 7 were met and that the licensee used realistic mathematical models to represent the 3D dynamic characteristics of the building structures for seismic response calculations in accordance with ASME/ANS Standard SFR-C1, C2, C5, and C6 requirements.

Deviation(s) or deficiency(ies) and Resolution: None

Consequence(s): N/A

The NRC staff concludes that:	
• The peer review findings have been addressed and the analysis approach has been accepted by the staff for the purposes of this evaluation. The peer review findings referred to relate to the SRs Seismic Fragility Analysis (SFR)-C1 through C6 in the ASME/ANS Standard, as well as to the requirements in the SPID.	Yes
• Although some peer review findings have not been resolved, the analysis is acceptable on another justified basis.	N/A
• The licensee's structural model meets the intent of the SPID guidance.	Yes
• The licensee's structural model does not meet the intent of the SPID guidance but is acceptable on another justified basis.	N/A

TOPIC 5: Use of Fixed-Based Dynamic Seismic Analysis of Structures for Sites Previously Defined as "Rock" (SPID Section 6.3.3)

<table>
<tr><td>Fixed-based dynamic seismic analysis of structures was used, for sites previously defined as "rock."

If <u>no</u>, this issue is moot.

If yes, on which structure(s)?

<u>Structure #1</u>:
If used, is V_S > about 5000 feet /second (ft./sec.)?

If 3500 ft./sec. < V_S < 5000 ft./sec., was peak-broadening or peak shifting used?

<u>Potential Staff Finding</u>:
The demonstration of the appropriateness of using this approach is adequate.</td><td>No

N/A

N/A

N/A</td></tr>
<tr><td colspan="2">Notes from staff reviewer:
The Sequoyah SPRA submittal states in Section 4.3.1 that the site is characterized as a firm rock site; however, soil structure interaction (SSI) analysis was performed for all major structures analyzed for the SPRA. Fixed-base analysis was not used for the seismic response analysis. The submittal stated that fixed-base analyses were only used for verification of some of the SSI models.

Deviation(s) or deficiency(ies) and Resolution: None.

Consequence(s): N/A</td></tr>
<tr><td>The NRC staff concludes that:
The peer review findings have been addressed and the analysis approach has been accepted by the staff for the purposes of this evaluation. The peer review findings referred to relate to the requirements in the SPID. No requirements in the ASME/ANS Standard specifically address this topic.
Although some peer review findings have not been resolved, the analysis is acceptable on another justified basis.
The licensee's use of fixed-based dynamic analysis of structures for a site previously defined as "rock" adequately meets the intent of the SPID guidance.
The licensee's use of fixed-based dynamic analysis of structures for a site previously defined as "rock" does not meet</td><td>N/A

N/A

N/A

N/A</td></tr>
</table>

the intent of the SPID guidance but is acceptable on another justified basis.	

TOPIC 6: Use of Seismic Response Scaling (SPID Section 6.3.2)

Seismic response scaling was used.	No
<u>Potential Staff Findings</u>: If a new Uniform Hazard Spectrum/Review Level Earthquake (UHS/RLE) is used, the shape is approximately like the spectral shape previously used for ISRS generation. If the shape is not similar, the justification for seismic response scaling is adequate. Consideration of non-linear effects is adequate.	 N/A N/A N/A
Notes from staff reviewer: Based on the staff's audit review, scaling of ground motion to generate ISRS was not used in the Sequoyah SPRA. Deviation(s) or deficiency(ies) and Resolution: None. Consequence(s): N/A	
The NRC staff concludes that: • The peer review findings have been addressed and the analysis approach has been accepted by the staff for the purposes of this evaluation. The peer review findings referred to relate to the SR SFR-C3 in the ASME/ANS Standard, as well as to the requirements in the SPID. • Although some peer review findings have not been resolved, the analysis is acceptable on another justified basis. • The licensee's use of seismic response scaling adequately meets the intent of the SPID guidance. • The licensee's use of seismic response scaling does not meet the intent of the SPID guidance but is acceptable on another justified basis.	 N/A N/A N/A N/A

TOPIC 7: Use of New Response Analysis for Building Response, ISRS, and Fragilities

The SPID does not provide specific guidance on performing new response analysis for use in developing ISRS and fragilities. The new response analysis is generally conducted when the criteria for use of existing models are not met or more realistic estimates are deemed necessary. The requirements for new analysis are included in the ASME/ANS Standard. See SRs SFR-C2, C4, C5, and C6. One of the key areas of review is consistency between the hazard and response analyses. Specifically, this means that there must be consistency among the ground motion equations, the SSI analysis (for soil sites), the analysis of how the seismic energy enters the base level of a given building, and the in-structure-response-spectrum analysis. Said another way, an acceptable SPRA must use these analysis pieces together in a consistent way. The following are high-level key elements that should have been considered:	
1. Foundation Input Response Spectra (FIRS) site response developed with appropriate building specific soil velocity profiles. Structure #1: Auxiliary-Control Building (ACB) Structure #2: Reactor Building (RB) Structure #3: Essential Raw Cooling Water (ERCW) Pumping Station Structure #4: Additional Equipment Buildings (AEBs) Structure #5: East Steam Valve Room (ESVR) Structure #6: Diesel Generator Building (DGB) Structure #7: Additional Diesel Generator Building (ADGB) Are all structures appropriately considered?	 Yes Yes Yes Yes Yes Yes Yes Yes
2. Are models adequate to provide realistic structural loads and response spectra for use in the SPRA? • Is the SSI analysis capable of capturing uncertainties and realistic? • Is the probabilistic response analysis capable of providing the full distribution of the responses?	Yes Yes Yes

Notes from staff reviewer:

The Sequoyah SPRA submittal describes in Section 4.3 the structural response analysis including SSI analysis that was used to develop the ISRS. The licensee used a probabilistic response analysis accounting for variabilities in strain-compatible soil profiles, structural characteristics, and the earthquake acceleration time histories performed using a Latin Hypercube Sampling (LHS) method. The SPRA report states that the LHS process includes generating 30 equal-probability bins with a parameter value sampled from each bin for each random variable to generate inputs for SSI simulations. The data for the LHS simulation was developed from 30 acceleration time history input ground motions that spectrally matched the FIRS of the building; 30 soil structure interaction simulations; and a combination of 30 structural stiffness and damping parameters spanning the proper range of statistical variation of these parameters.

The Sequoyah SSI analysis is based on a sub-structuring approach and used an industry standard software for model development and analysis, where the structural systems were modeled using 3D finite element or lumped-mass-stick models and the soil as a horizontal layer. The probabilistic SSI analysis includes consideration of hazard compatible soil properties, ground motion incoherency, simultaneous analysis in the three spatial directions, and a cut-off frequency of 50 Hz.

The probabilistic structural response was used to calculate ISRS and the foundation and building displacements. The ISRS was calculated, for a range of damping values, from acceleration time history output generating 50th median (50th percentile) and conservative (84th percentile) spectral accelerations at selected locations.

There are three F&Os based on peer review findings related to SRs SFR-C2, C4, C5 and C6. F&O 3-4 (SFR-C6) commented that the structural response analysis of the DGB should be performed for a range of ground motions of interest for the SPRA and the soil strain levels should correspond to the input ground motion. In response, the licensee evaluated the seismic response of the DGB at higher ground motions considering the non-linear hysteretic behavior of the soil for the SSI analysis. The licensee concluded that, at the frequency range of interest for DGB equipment, the ISRS generated based on SSI using equivalent-linear soil properties compatible with the GMRS ground motion was conservative compared to the SSI with non-linear soil properties and higher ground motion, thus have minimal effect on the risk results.

F&O 5-3 (SFR-C4) commented that because of structural gaps in the NSSS components, the non-linear dynamic response was not adequately considered for higher ground motions. The licensee's reevaluation showed that the dynamic response was not affected by higher ground motion magnitude and the effects of varying non-linear support conditions on seismic demand was negligible. Thus, no update was required.

In response to F&O 5-6 (SFR-C5) recommending reassessment of application of clipping of ISRS, the licensee reevaluated clipping using EPRI NP-6041 and TR-103959 addressing the issues with double peaks. The ISRS were updated along with the fragility values and final risk quantification.

The peer review F&Os related to the structural response analysis have been resolved

through the NRC-approved independent assessment process documented in NEI 12-13, Appendix X. Based on the NRC staff's review of information in the submittal and auditing of structural response documentation via the eportal, the staff finds the probabilistic approach to evaluate structural response and floor response spectra to be appropriate. The probabilistic simulation approach, consideration of variability in soil and structural properties, and the number of simulations used are consistent with ASCE-4 recommendations and industry practice. Deviation(s) or deficiency(ies) and Resolution: None. Consequence(s): N/A	
The NRC staff concludes that:	
• The peer review findings have been addressed and the analysis approach has been accepted by the staff for the purposes of this evaluation. The peer review findings referred to relate to the SRs SFR-C2, C4, C5, and C6 in the ASME/ANS Standard, as well as to the requirements in the SPID.	Yes
• Although some peer review findings have not been resolved, the analysis is acceptable on another justified basis.	N/A
• The licensee's FIRS modeling is consistent with the prior NRC review of the GMRS and soil velocity information.	Yes
• The licensee's structural model meets the intent of the SPID guidance and the ASME/ANS Standard's requirements.	Yes
• The response analysis accounts for uncertainties in accordance with the SPID guidance and the ASME/ANS Standard's requirements.	Yes
• The NRC staff concludes that an acceptable consistency has been achieved among the various analysis pieces of the overall analysis of site response and structural response.	Yes
• The licensee's structural model does not meet the intent of the SPID guidance and the ASME/ANS Standard's requirements but is acceptable on another justified basis.	N/A

TOPIC 8: Screening by Capacity to Select SSCs for Seismic Fragility Analysis (SPID Section 6.4.3)

The selection of SSCs for seismic fragility analysis used a screening approach by capacity following Section 6.4.3 of the SPID. If no, see items D and E. If yes, see items A, B, and C.	Yes
Potential Staff Findings:	
A) The recommendations in Section 6.4.3 of the SPID were followed for the screening aspect of the analysis, using the screening criteria therein.	Yes
B) The approach for retaining certain SSCs in the model with a screening-level seismic capacity follows the recommendations in Section 6.4.3 of the SPID and has been appropriately justified.	Yes
C) The approach for screening out certain SSCs from the model based on their inherent seismic ruggedness follows the recommendations in Section 6.4.3 of the SPID and has been appropriately justified.	Yes
D) The ASME/ANS Standard has been followed.	N/A
E) An alternative method has been used and its use has been appropriately justified.	N/A
Notes from staff reviewer: Section 4.4.1 of the Sequoyah SPRA report states that a multi-step screening process was used to reduce the SEL to the point where only significant SSCs were included in the SPRA. Some systems, as identified in Table 4.1-1 of the SPRA report, and those located outside Category I structures were initially screened out because they were judged to be non-contributors to the overall risk. In Section 4.4.1 of the submittal, Sequoyah discussed screening of inherently rugged SSCs and rule-of-the-box components. Fragility of valves that change state or are required to change state during or after a seismic event are included in the SPRA. Using the audit process, the NRC staff confirmed that the capacity-based screening criterion recommended in SPID Section 6.4.3 was not used to screen out high capacity components at Sequoyah. Instead, the licensee used a more conservative approach by retaining the high capacity components in the SPRA quantification model with an assigned high median capacity-based on walkdown insights.	

Section 4.1.2 of the Sequoyah SPRA report states that a number of relay/breaker chatter scenarios were screened out from further evaluation based on no impact to component function resulting from the seismic event. This screening process for components sensitive to high frequency is discussed in further detail under Topics #10 and #11 of this checklist. In response to F&O 7-5, the temperature sensors that were screened out as rugged were reevaluated. The temperature sensors were re-screened because they were mounted on block walls. The screening of components of high-seismic capacity was addressed in accordance with ASME/ANS Standard SFR-B1 requirements. Deviation(s) or deficiency(ies) and Resolution: None. Consequence(s): N/A	
The NRC staff concludes that:	
• The peer review findings have been addressed and the analysis approach has been accepted by the staff for the purposes of this evaluation. The peer review findings referred to relate to the SR SFR-B1 in the ASME/ANS Standard, as well as to the requirements in the SPID.	Yes
• Although some peer review findings have not been resolved, the analysis is acceptable on another justified basis.	N/A
• The licensee's use of a screening approach for selecting SSCs for fragility analysis meets the intent of the SPID guidance.	Yes
• The licensee's use of a screening approach for selecting SSCs for fragility analysis does not meet the intent of the SPID guidance but is acceptable on another justified basis.	N/A

TOPIC 9: Use of the CDFM/Hybrid Methodology for Fragility Analysis (SPID Section 6.4.1)

The CDFM/Hybrid method was used for seismic fragility analysis. If no, See item C) below and next issue. If yes: Potential Staff Findings:	Yes
A) The recommendations in Section 6.4.1 of the SPID were followed appropriately for developing the CDFM High Confidence Low Probability of Failure (HCLPF) capacities.	Yes
B) The Hybrid methodology in Section 6.4.1 and Table 6-2 of the SPID was used appropriately for developing the full seismic fragility curves.	Yes
C) An alternative method has been used appropriately for developing full seismic fragility curves.	N/A

Notes from staff reviewer:

Section 4.4.2 of the Sequoyah SPRA report states that three quantification processes were used for fragility evaluation of SEL components, starting with representative fragilities in the first quantification and improving the fragilities for top risk contributors to SCDF and SLERF in the second and third quantifications. During the second and third risk quantifications, the fragility for some of the risk-significant items was refined using the CDFM/Hybrid methodology based on EPRI NP-6041 and other EPRI guidance, to develop the HCLPF. The licensee stated in the submittal that the median capacity in the hybrid approach was developed by combining the HCLPF with the generic variabilities given in Table 6.2 of the SPID. Tables 5.4-3, 5.4-4, 5.5-3, and 5.5-4 show that the fragility of most top risk contributors was determined using the hybrid method. During the audit of Sequoyah's supporting documentation, the staff reviewed the fragility of selected components and confirmed that the CDFM/Hybrid methodology followed standard practice using site-specific data and ISRS developed from structural response analysis. Failure modes considered included structural failure, functional failures, seismic interaction with surrounding equipment or objects, local failure modes due to soil failure, and breaker trip or relay chatter. In addition, the licensee stated in the submittal that the first quantification of fragility was also based on estimating HCLPF and used generic uncertainty values from the SPID.

During the audit process, the staff noted that fragilities for some SSCs were based on fragility evaluation from the Watts Bar SPRA. The licensee provided justification for the assumed capacity based on identical SSCs at both facilities. The demand was based on a comparison of the spectral accelerations at the two facilities at frequencies associated with the SSC function and its mounting location. For the fragility of NSSS components, the licensee performed a sensitivity study (Case #25 documented in Table 5.7-1 of the submittal). The sensitivity analysis was used to justify fragility evaluation of NSSS components using the non-conservative comparison of the 1 percent damped SSE spectrum at Watts Bar and the 5 percent damped reference level earthquake spectrum at Sequoyah. The sensitivity analysis shows small to moderate reduction in SCDF and SLERF, and the approach to fragility evaluation of NSSS components is acceptable for this submittal.

During the audit process, the NRC staff noted that the structural fragility of buildings developed for the Watts Bar SPRA was used for Sequoyah by implementing a scaling method. The fragility was modified by using the ratio between GMRS of the two sites at critical frequencies. The licensee clarified that GMRS for both the Watts Bar and Sequoyah facilities are 5 percent damped spectra. The staff found the scaling to be appropriate.

The staff also confirmed that the procedure for development of CDFM/Hybrid fragilities using EPRI NP-6041-SL, EPRI 1019200, EPRI TR-103959 and other updated EPRI guidance is consistent with the recommendation in Section 6.4.1 of the SPID.

Deviation(s) or deficiency(ies) and Resolution: None.

Consequence(s): N/A

The NRC staff concludes that:	
• The peer review findings have been addressed and the analysis approach has been accepted by the staff for the purposes of this evaluation. The peer review findings referred to relate to the requirements in the SPID. No requirements in the ASME/ANS Standard specifically address this Topic.	Yes
• Although some peer review findings have not been resolved, the analysis is acceptable on another justified basis.	N/A
• The licensee's use of the CDFM/Hybrid method for seismic fragility analysis meets the intent of the SPID guidance.	Yes
• The licensee's use of the CDFM/Hybrid method for seismic fragility analysis does not meet the intent of the SPID guidance but is acceptable on another justified basis.	N/A

TOPIC 10: Capacities of SSCs Sensitive to High-Frequencies (SPID Section 6.4.2)

The SPID requires that certain SSCs that are sensitive to high-frequency seismic motion must be analyzed in the SPRA for their seismic fragility using a methodology described in Section 6.4.2 of the SPID.	Yes
Potential Staff Findings:	
The NRC staff review of the SPRA's fragility analysis of SSCs sensitive to high frequency seismic motion finds that the analysis is acceptable.	Yes
The flow chart in Figure 6-7 of the SPID was followed.	Yes
The flow chart was not followed but the analysis is acceptable on another justified basis.	N/A
Notes from staff reviewer: Section 4.1.2 of the Sequoyah submittal states that the devices potentially sensitive to high-frequency seismic motion were included in the SPRA. The NRC staff confirmed that the list of SSCs sensitive to high frequencies was based on information in EPRI 30020002997. The NRC staff's review of supporting documents during the audit process showed that high capacity components were screened out consistent with Figure 6-7 of the SPID. Some medium and low capacity components were also screened out based on the circuit analysis. The SSCs sensitive to high frequencies are included in the SPRA based on a capacity versus demand evaluation consistent with the recommendation in Section 6.4.2 of the SPID. Deviation(s) or deficiency(ies) and Resolution: None. Consequence(s): N/A	
The NRC staff concludes that:	
• The peer review findings have been addressed and the analysis approach has been accepted by the staff for the purposes of this evaluation. The peer review findings referred to relate to the SR SFR-F3 in the ASME/ANS Standard, as well as to the requirements in the SPID.	Yes
• Although some peer review findings have not been resolved, the analysis is acceptable on another justified basis.	N/A

• The licensee's fragility analysis of SSCs sensitive to high frequency seismic motion meets the intent of the SPID guidance. • The licensee's fragility analysis of SSCs sensitive to high-frequency motion does not meet the intent of the SPID guidance but is acceptable on another justified basis.	Yes N/A

TOPIC 11: Capacities of Relays Sensitive to High-Frequencies (SPID Section 6.4.2)

The SPID requires that certain relays and related devices (generically, "relays") that are sensitive to high-frequency seismic motion must be analyzed in the SPRA for their seismic fragility. Although following the ASME/ANS Standard is generally acceptable for the fragility analysis of these components, the SPID (Section 6.4.2) contains additional guidance when either circuit analysis or operator-action analysis is used as part of the SPRA to understand a given relay's role in plant safety. When one or both are used, the NRC reviewer should use the following elements of the checklist.	
i) <u>Circuit analysis</u>: The seismic relay-chatter analysis of some relays relies on circuit analysis to assure that safety is maintained. (A) If <u>no</u>, then (B) is moot. (B) If <u>yes:</u> <u>Potential Staff Finding</u>: The approach to circuit analysis for maintaining safety after seismic relay chatter is acceptable.	Yes Yes
ii) <u>Operator actions</u>: The relay-chatter analysis of some relays relies on operator actions to assure that safety is maintained. (A) If <u>no</u>, then (B) is moot. (B) If <u>yes:</u> <u>Potential Staff Finding</u>: The approach to analyzing operator actions for maintaining safety after seismic relay chatter is acceptable.	Yes Yes
Notes from staff reviewer: Section 4.1.2 of the Sequoyah SPRA submittal states that relay and breaker chatter evaluation was performed in accordance with the requirements in the ASME/ANS Standard and SPID Section 6.4.2 requirements. During the audit of supporting documentation, the NRC staff confirmed that the circuit analysis was part of the chatter analysis evaluation performed for Sequoyah. The circuit analysis resulted in many relay	

<table>
<tr><td colspan="2">chatter scenarios being screened from further evaluation based on having no impact on component function. For the chatter analysis, relays were assumed to malfunction only during the strong motion portion of the seismic event.

The relays that were screened in for further evaluation are summarized in Table 4.1-2 of the SPRA report. In most cases the relays were modeled as fragility groups, while in some cases they were included in other models.

Section 4.1.2 of the Sequoyah SPRA report states that operator recovery actions were credited in the SPRA model. The effects of relay chatter were addressed in accordance with ASME/ANS Standard SPR-B6 requirements.

Deviation(s) or deficiency(ies) and Resolution: None.

Consequence(s): N/A</td></tr>
<tr><td>The NRC staff concludes that:

• The peer review findings have been addressed and the analysis approach has been accepted by the staff for the purposes of this evaluation. The peer review findings referred to relate to the SRs Seismic Plant Response Analysis (SPR)-B6 (Addendum A) or SPR-B4 (Addendum B) in the ASME/ANS Standard, as well as to the requirements in the SPID.

• Although some peer review findings have not been resolved, the analysis is acceptable on another justified basis.

• The licensee’s analysis of seismic relay-chatter effects meets the intent of the SPID guidance.

• The licensee’s analysis of seismic relay-chatter effects does not meet the intent of the SPID guidance but is acceptable on another justified basis.</td><td>Yes

N/A

Yes

N/A</td></tr>
</table>

TOPIC 12: Selection of Dominant Risk Contributors that Require Fragility Analysis Using the Separation of Variables (SOV) Methodology (SPID Section 6.4.1)

The CDFM methodology has been used in the SPRA for analysis of the bulk of the SSCs requiring seismic fragility analysis.	Yes
If no, the staff review will concentrate on how the fragility analysis was performed, to support one or the other of the “potential staff findings” noted just below.	N/A
If yes, significant risk contributors for which use of SOV fragility calculations would make a significant difference in the SPRA results have been selected for SOV calculations.	Yes
Potential Staff Findings: A) The recommendations in Section 6.4.1 of the SPID were followed concerning the selection of the “dominant risk contributors” that require additional seismic fragility analysis using the SOV methodology.	Yes
B) The recommendations in Section 6.4.1 were not followed, but the analysis is acceptable on another justified basis.	N/A
Notes from staff reviewer: Sections 4.4.2 and 4.4.3 of the SPRA submittal explain that a three-step quantification approach was performed in which risk quantification and sensitivity studies were used to identify the important risk contributors. After the important risk contributors were identified, the fragility analyses for these contributors were refined to reduce conservatism using the CDFM-based Hybrid Method, and then if needed, the SOV approach. Significant contributors to the seismic risk were evaluated using SOV procedures consistent with the requirements of SPID Section 6.4.1. The NRC staff notes that Tables 5.4-3, 5.4-4, 5.5-3, and 5.5-4 of the SPRA submittal, which provide the SCDF and SLERF importance measures, indicate that the majority of fragility groups were analyzed using the CDFM method. For top risk contributors (e.g., buried ERCW piping, Relay Chatter Solid-State Protection System (SSPS) Cabinets, and the Refueling Water Storage Tank), the SOV method was used for fragility evaluation. After the final quantification, many of the SSCs with fragilities calculated using the SOV approach dropped off the list of important risk contributors. The submittal, supported by information provided in response to NRC audit questions, explains that the first quantification was based on site-specific representative fragilities that were typically developed using simplified methods (e.g., scaling, screening). The NRC staff notes that among the top risk contributors, only the loss of offsite power	

(LOSP) fragility group was analyzed using a representative fragility. The LOSP fragility group is based on a generic fragility that includes the switchyard and the grid outside the plant boundary, which limits the ability of the licensee to improve the fragility estimate or make plant improvements to this fragility group. During the third risk quantification and the reassessment following the peer review, the licensee developed the fragility of relays in the SSPS cabinets (Fragility Group SEIS_0-30-1) using the SOV approach. Functional fragility was based on available component specific test data and Seismic Qualification Utility Group (SQUG) generic equipment ruggedness spectra (GERS). Horizontal and vertical amplification factors for the relay specific locations were applied to the set of median and 84 percent clipped ISRS at the cabinet locations. The staff finds the relay fragility was developed using an approach that is consistent with SPID recommendation and in accordance with Section 3 of EPRI TR-103959, including updates from EPRI 1002988 and EPRI 1019200. In response to F&Os 3-10 and 5-2, the licensee developed a fragility value for the buried ERCW piping. The pipe material strain resulting from localized soil deformation caused by vertical settlement and lateral spreading of the ground was evaluated for varying ground motions. The fragility parameters, median capacity and composite uncertainty, were estimated based on median strain of the pipe exceeding the 5 percent level. The fragility of ERCW 6-inch diameter corroded pipe in the DGB/AB area was the governing fragility when compared to other higher diameter pipes and was used in the SPRA analysis. The submittal included the results of various sensitivity studies to investigate the impact of increased fragilities. The results did not show major impacts to the SCDF or SLERF. Therefore, additional refinement to the fragilities is not expected to alter the contributors to the seismic risk. The NRC staff finds that the CDFM method was used in the seismic PRA for analysis of the bulk of the SSCs requiring seismic fragility analysis and that the SOV approach was used for SSCs that are significant contributors to the seismic risk, consistent with the requirements in SPID section 6.4.1. Deviation(s) or deficiency(ies) and Resolution: None. Consequence(s): N/A	
The NRC staff concludes that: • The peer review findings have been addressed and the analysis approach has been accepted by the staff for the purposes of this evaluation. The peer review findings referred to relate to the requirements in the SPID. No requirements in the ASME/ANS Standard specifically address this Topic.	Yes

• Although some peer review findings have not been resolved, the analysis is acceptable on another justified basis.	N/A
• The licensee's method for selecting the "dominant risk contributors" for further seismic fragilities analysis using the SOV methodology meets the intent of the SPID guidance.	Yes
• The licensee's method for selecting the "dominant risk contributors" for further seismic fragilities analysis using the SOV methodology does not meet the intent of the SPID guidance but is acceptable on another justified basis.	N/A

TOPIC 13: Evaluation of SLERF (SPID Section 6.5.1)

The NRC staff review of the SPRA's analysis of SLERF finds an acceptable demonstration of its adequacy. Potential Staff Findings: A) The analysis follows each of the elements of guidance for SLERF analysis in Section 6.5.1 of the SPID, including in Table 6-3. B) The SLERF analysis does not follow the guidance in Table 6-3 but the analysis is acceptable on another justified basis.	Yes Yes N/A
Notes from staff reviewer: Section 4.1 of the submittal includes a list of systems considered for the SEL. Systems associated with important containment functions, including the ice condenser, containment spray, containment heat removal, containment isolation, and hydrogen mitigation are included in the SEL. In accordance with the SPID, the internal events PRA (IEPRA) was adapted to incorporate seismic events. Section 5.1.5 of the SPRA report states that the complete level 2 model developed for the IEPRA was used for the SPRA since the event progression following core melt is similar. Tables 5.5-3 and 5.5-4 of the submittal present the fragility groups with the most significant failure contribution to SLERF. Tables 5.5-7 and 5.5-8 show that the most significant accident sequences all include failure of important containment functions (e.g., failure of containment isolation of line greater than 2 inches, unavailability of the containment air return fans, failure of hydrogen igniters, and failure of containment heat removal). There were no LERF-related F&Os that remain unresolved for this submittal. (Resolution of F&Os is discussed in more detail under Topic #14.) Deviation(s) or deficiency(ies) and Resolution: None Consequence(s): N/A	
The NRC staff concludes that: • The peer review findings have been addressed and the analysis approach has been accepted by the staff for the purposes of this evaluation. The peer review findings referred to relate to SRs SFR-F4, SPR-E1, SPR-E2, and SPR-E6 (Addendum B only) in the ASME/ANS Standard, as well as to the requirements in the SPID. • Although some peer review findings have not been resolved, the analysis is acceptable on another justified basis.	 Yes N/A

• The licensee's analysis of SLERF meets the intent of the SPID guidance.	Yes
• The licensee's analysis of SLERF does not meet the intent of the SPID guidance but is acceptable on another justified basis.	N/A

TOPIC 14: Peer Review of the SPRA, Accounting for NEI 12-13 (SPID Section 6.7)

The NRC staff review of the SPRA's peer review findings, observations, and their resolution finds an acceptable demonstration of the peer review's adequacy. Potential Staff Findings: A) The analysis follows each of the elements of the peer review guidance in Section 6.7 of the SPID. B) The composition of the peer review team meets the SPID guidance. C) The peer reviewers focusing on seismic response and fragility analysis have successfully completed the Seismic Qualifications Utility Group training course or equivalent (see SPID Section 6.7). In what follows, a distinction is made between an "in-process" peer review and an "end-of-process" peer review of the completed SPRA submittal. If an in-process peer review is used, go to (D) and then skip (E). If an end-of-process peer review is used, skip (D) and go to (E). D) The "in-process" peer review process followed the guidance in the SPID (Section 6.7), including the three "bullets" and the guidance related to NRC's additional input in the paragraph immediately following those three bullets. These three bullets are: • The SPRA findings should be based on a consensus process, and not based on a single peer review team member • A final review by the entire peer review team must occur after the completion of the SPRA project • An "in-process" peer review must assure that peer reviewers remain independent throughout the SPRA development activity.	Yes Yes Yes Yes N/A
If no, go to (F). If yes, the "in-process" peer review approach is acceptable. Go to (G).	

E) The “end-of-process” peer review process followed the peer review guidance in the SPID (Section 6.7). If no, go to (F). If yes, the “end-of-process” peer review approach is acceptable. Go to (G).	Yes
F) The peer-review process does not follow the guidance in the SPID but is acceptable on another justified basis.	N/A
G) The licensee peer-review findings were satisfactorily resolved or were determined not to be significant to the SPRA conclusions for this evaluation.	Yes

Notes from staff reviewer:

The SPRA submittal follows the recommendations of Section 6.7 of the SPID. Section 5.2 and Appendix A.2 of the SPRA submittal describe the peer review process used to establish the technical adequacy of the SPRA. The technical adequacy of the IEPRA that provides the foundation for the SPRA is described in Appendix A.7 of the SPRA submittal. The staff found the IEPRA to be technically acceptable for the licensee to implement 10 CFR 50.69 as documented in its safety evaluation dated September 18, 2019 (ADAMS Accession No. ML19179A135). This safety evaluation and the associated license amendment request dated March 16, 2018 (ADAMS Accession No. ML18075A365), which indicated that the IEPRA was reviewed to identify PRA upgrades as defined by the ASME/ANS RA-S-2009 PRA standard, were used to inform this review of the Sequoyah SPRA submittal.

A full-scope SPRA peer review, consistent with the guidance in Regulatory Guide (RG) 1.200, Revision 2 and Section 1-6 of ASME/ANS RA-Sb-2013 PRA Standard, was conducted in April 2018. This peer review assessed the technical adequacy of the Sequoyah SPRA against the Capability Category (CC) II Supporting Requirements (SRs) of ASME/ANS RA-Sb-2013 PRA Standard, consistent with Tables 6-4 through 6-6 of the SPID. The review was conducted using the peer review process defined in NEI 12-13. The submittal describes the qualifications of the eight reviewers. Walkdowns were performed “by several members of the peer review team” who have the appropriate SQUG training. The full-scope SPRA peer review performed in 2018 resulted in 54 Finding-level F&Os.

An F&O closure review was performed in February 2019 using the independent assessment process outlined in Appendix X to NEI 12-13. The NRC staff’s review of the F&O report during the audit found that the closure review was performed in accordance with the conditions specified in the NRC acceptance letter regarding the Appendix X process dated May 3, 2017 (ADAMS Accession No. ML17079A427). The F&O closure team, consisting of six members, reviewed the dispositions to the finding-level F&Os from the April 2018 full-scope SPRA peer review. The SPRA submittal states, and NRC staff confirmed, that all F&Os were determined to be Met or Met at CC II and were closed. The NRC staff reviewed the SPRA F&O dispositions included with the submittal and did not identify any modeling issues that would be expected to impact the conclusions.

The IEPRA and internal flooding PRA were used as the basis for the development of the SPRA. A full-scope IEPRA peer review was conducted in January 2011 by the Pressurized Water Reactor Owners Group (PWROG) against the CC II SRs of the ASME/ANS PRA Standard RA-S-2009 and RG 1.200, Revision 2. Subsequent to that peer review an F&O closure review was performed in May 2017 using the independent assessment process outlined in Appendix X to NEI 05-04. The F&O closure review was performed against the ASME/ANS PRA Standard RA-S-2009 and RG 1.200, Revision 2. The NRC safety evaluation of the licensee's request to adopt 10 CFR 50.69 (ADAMS Accession No. ML19179A135) concluded that the Sequoyah internal events F&O closure review was performed in accordance with the conditions specified in the NRC acceptance letter regarding the Appendix X process. The SPRA submittal presents the F&O closure review results in Table A-3 of Appendix A and shows that all 31 Finding-level F&Os from the full-scope IEPRA peer review have been closed. The NRC safety evaluation of the licensee's request to adopt 10 CFR 50.69 concludes that ASME/ANS RA-SA-2009 PRA Standard Supporting Requirements associated with the closed F&Os meet CC II. The NRC staff reviewed the IEPRA F&O dispositions and did not identify any modeling issues that could impact the conclusions of the SPRA report. Deviation(s) or deficiency(ies) and Resolution: None Consequence(s): N/A

The NRC staff concludes that:	
• The licensee's peer-review process meets the intent of the SPID guidance.	Yes
• The licensee's peer-review process does not meet the intent of the SPID guidance but is acceptable on another justified basis.	N/A

TOPIC 15: Documentation of the SPRA (SPID Section 6.8)

The NRC staff review of the SPRA's documentation as submitted finds an acceptable demonstration of its adequacy. The documentation should include all the items of specific information contained in the 50.54(f) letter as described in Section 6.8 of the SPID.	Yes Yes
Notes from staff reviewer: The Sequoyah SPRA submittal follows the recommendations in Section 6.8 of the SPID. Tables 2-1 and 2-2 of the report provide a cross-reference of information requested by the 10 CFR 50.54(f) letter and additional information specified in SPID Section 6.8, respectively, to the corresponding SPRA report sections. The information provided is sufficiently detailed to allow the NRC staff to assess the results of all key aspects of the analysis. Sections 3.1 and 5.3.2 of the submittal identify and discuss key assumptions and sources of model uncertainty for the SPRA. Sections 5.4 and 5.5 of the submittal include the SCDF and SLERF quantification results. Section 5.6 of the submittal presents the parametric data uncertainty analysis results including the total SCDF and SLERF point estimates, mean values, 5th percentiles, median values, and 95th percentiles. Section 5.7 of the submittal provides the results of 31 sensitivity studies. The submittal does not refer to or describe pertinent information from the site's Individual Plant Examination of External Events (IPEEE) program as suggested in the SPID (e.g., all functional/systemic event trees); however, the SPID only identifies this IPEEE information as guidance for consideration in the 50.54(f) response. Section 6.8 of the SPID states that the submittal should provide sufficient information for the NRC staff to assess the sensitivity of the SPRA results to all key aspects of the analysis. Table 5.7-1 of the SPRA report provides the results of 31 sensitivity studies. Section 5.1 of the submittal states that the permanently installed 480V Flexible and Diverse Coping Strategies (FLEX) diesel generators and the 6.9kV FLEX diesel generators are credited in the SPRA model. Section 5.4.4.4 of the submittal clarifies that operator actions to activate these diesel generators are credited in the SPRA, but operator actions associated with the portable FLEX equipment were assumed to be failed. The licensee performed a sensitivity study and presented the results in Table 5.7-1 (Case #3), which shows that credit for the permanently installed FLEX diesel generators has a small impact on the SCDF and SLERF results. Therefore, NRC staff finds that uncertainties related to operator actions for portable FLEX equipment have a minimal impact on the conclusions of the submittal. Additional sensitivity studies are discussed in Enclosure 2. Deviation(s) or deficiency(ies) and Resolution: None Consequence(s): N/A	
The NRC staff concludes that:	

• The licensee's documentation meets the intent of the SPID guidance. The documentation requirements in the ASME/ANS Standard can be found in HLR-SHA-J, HLR-SFR-G, and HLR-SPR-F.	Yes
• The licensee's documentation does not meet the intent of the SPID guidance but is acceptable on another justified basis.	N/A

Topic 16: Review of Plant Modifications and Licensee Actions, If Any

<table>
<tr><td>The licensee:
identified modifications necessary to achieve seismic risk improvements.
provided a schedule to implement such modifications (if any), consistent with the intent of the guidance
provided Regulatory Commitment to complete modifications
provided Regulatory Commitment to report completion of modifications.</td><td>No

N/A

N/A

N/A</td></tr>
<tr><td>Plant will:
complete modifications by:
report completion of modifications by:</td><td>N/A

N/A</td></tr>
<tr><td colspan="2">Notes from the Reviewer:

NRC staff's identification, review, and conclusion on potential plant modification is discussed in Enclosure 2.

Deviation(s) or Deficiency(ies), and Resolution: None

Consequences: N/A</td></tr>
<tr><td>The NRC staff concludes that:
The licensee identified plant modifications necessary to achieve the appropriate risk profile.
The licensee provided a schedule to implement the modifications (if any) with appropriate consideration of plant risk and outage scheduling.</td><td>No

N/A</td></tr>
</table>

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NRC, 2017a: Giitter, Joseph, and Ross-Lee, Mary Jane, U.S. Nuclear Regulatory Commission, letter to Krueger, Greg, Nuclear Energy Institute, "U.S. Nuclear Regulatory Commission Acceptance on Nuclear Energy Institute Appendix X to Guidance 05-04, 07-12, and 12-13, Close-Out of Facts and Observations (F&Os)," dated May 3, 2017 (ADAMS Accession No. ML17079A427).

NRC Staff SPRA Submittal Detailed Screening Evaluation

Introduction

The Sequoyah Nuclear Plant, Units 1 and 2 (Sequoyah) Seismic Probabilistic Risk Assessment (SPRA) submittal (Agencywide Documents Access and Management System (ADAMS) Accession No. ML19291A003) indicates that the mean seismic core damage frequency (SCDF) is 1.3E-05 per reactor-year (/rx-yr) for Unit 1 and 1.5E-05/rx-yr for Unit 2, and the mean seismic large early release frequency (SLERF) is 6.9E-06/rx-yr for both Units 1 and 2. The NRC staff compared these values against the guidance in NRC staff memorandum dated August 29, 2017 (ADAMS Accession No. ML17146A200), titled, "Guidance for Determination of Appropriate Regulatory Action Based on Seismic Probabilistic Risk Assessment Submittals in Response to Near Term Task Force Recommendation 2.1: Seismic" (hereafter referred to as the SPRA Screening Guidance), which establishes a process the NRC staff uses to develop a recommendation on whether the plant should move forward as a Group 1, 2, or 3 plant.[1]

The SPRA screening guidance is based on NUREG/BR-0058, Revision 4, "Regulatory Analysis Guidelines of the U.S. Nuclear Regulatory Commission," (ADAMS Accession No. ML042820192), NUREG/BR-0184, "Regulatory Analysis Technical Evaluation Handbook," (ADAMS Accession No. ML050190193), and NUREG-1409, "Backfitting Guidelines," (ADAMS Accession No. ML032230247), as informed by Nuclear Energy Institute (NEI) guidance document NEI 05-01, "Severe Accident Mitigation Alternatives (SAMA) Analysis Guidance Document" (ADAMS Accession No. ML060530203). To determine the significance of proposed modifications in terms of safety improvement, NUREG/BR-0058 uses screening criteria based on the estimated reduction in core damage frequency, as well as the conditional probability of early containment failure or bypass. Per NUREG/BR-0058, the conditional probability of early containment failure or bypass is a measure of containment performance and the purpose of its inclusion in the screening criteria is to achieve a measure of balance between accident prevention and mitigation. The NUREG/BR-0058 uses a screening criterion of 0.1 or greater for conditional probability of early containment failure or bypass. In the context of the SPRA reviews, the staff guidance uses SCDF and SLERF as the screening criteria where SLERF is directly related to the conditional probability of early containment failure or bypass. Following NUREG/BR-0058, the threshold for the screening criterion in the staff guidance for SLERF is (1.0E-6/rx-yr), or 0.1 times the threshold for the screening criterion for SCDF (1.0E-5/rx-yr).

Because the SCDF and SLERF for Sequoyah Units 1 and 2 were above the initial screening values of 1.0E-5/rx-yr and 1.0E-6/rx-yr, respectively, the NRC staff performed a detailed screening following the SPRA Screening Guidance. The detailed screening results show that Sequoyah should be considered a Group 1 plant because:

- Sufficient reductions in SCDF and SLERF cannot be achieved by potential modifications considered in this evaluation to constitute cost-justified substantial safety improvements based upon importance measures, available information, and engineering judgement;

[1] The groups are defined as follows: regulatory action not warranted (termed Group 1), regulatory action should be considered (termed Group 2), and more thorough analysis is needed to determine if regulatory action should be considered (termed Group 3).

- Additional consideration of containment performance, as described in NUREG/BR-0058, does not identify a modification that would result in a substantial safety improvement; and

- The staff did not identify any potential modifications that would be appropriate to consider necessary for adequate protection or compliance with existing requirements.

As such, additional refined screening, or further evaluation, was not required.

Tennessee Valley Authority (TVA, the licensee), in performing its seismic analysis in response to the Near-Term Task Force Recommendation 2.1, and the NRC staff in conducting its review, did not identify concerns that would require action above and beyond existing regulations to maintain the level of protection necessary to avoid undue risk to public health and safety. In addition, there were no issues identified as non-compliances with the Sequoyah license, or with the rules and orders of the Commission. For these reasons, the licensee and the NRC staff did not identify a potential modification necessary for adequate protection or compliance with existing regulations.

<u>Detailed Screening</u>

The detailed screening uses information provided in the Sequoyah SPRA report, particularly the importance measures, SCDF, SLERF, and other information described below, to establish threshold and target values that are used to identify areas where potential cost-justified substantial safety improvements might exist. The detailed screening process makes several simplifying assumptions, similar to a Phase 1 severe accident mitigation alternative (SAMA) analysis (NEI 05-01, ADAMS Accession No. ML060530203) used for license renewal applications. The detailed screening process uses risk importance values as defined in NUREG/CR-3385, "Measures of Risk Importance and Their Applications" (ADAMS Accession No. ML071690031). The NUREG/CR-3385 states that the risk reduction worth (RRW) importance value is useful for prioritizing feature improvements that can most reduce the risk. The Sequoyah SPRA report provides Fussell-Vesely (F-V) importance values, which were converted to RRW values by the NRC staff for this screening evaluation using a standard relationship formulation. Data used to develop the maximum averted cost-risk (MACR) for the SAMA analysis provided in the *Applicant's Environmental Report, Operating License Renewal Stage, Sequoyah Nuclear Plant - License Renewal Application*, dated January 7, 2013 (ADAMS Accession Nos. ML13024A010, ML13024A011, and ML13024A012), was used to calculate the RRW threshold. For this analysis, the NRC staff determined the RRW threshold from the SCDF-based MACR to be 1.024 for both Units 1 and 2. The MACR calculation includes estimation of offsite exposures and offsite property damage, which captures the impact of SLERF. Therefore, separate SLERF-based MACR calculations were not performed. The target RRW value (as defined by the SPRA screening guidance) based on the mean and 95th percentile SCDF and SLERF were calculated by NRC staff to be between 4.70 and 1.08.

Section 5 of the Sequoyah SPRA submittal includes tables listing and describing the fragility groups for structures, systems, and components (SSCs) and the operator errors that are the most significant seismic failure contributors to SCDF and SLERF. The descriptions of the significant contributors included the F-V importance measures. The NRC staff utilized the F-V values to calculate the RRW, the maximum monetary value of eliminating the failure, and the contribution to SCDF or SLERF of each contributor. The results are provided in Tables 1 and 3 for the SCDF contributors for Units 1 and 2, respectively, and in Tables 2 and 4 for the SLERF contributors for Units 1 and 2, respectively, that have an RRW greater than about 1.005. These tables provide the following information by column: (1) fragility group/event, (2) description of the component, (3) failure mode, (4) RRW, and (5) maximum SCDF reduction (MCR) or SLERF reduction (MLR) from eliminating the failure.

Based on the F-V values provided in Sections 5.4 and 5.5 of the SPRA submittal, elimination or reduction in the contribution of the following fragility group failures and operator errors, individually or in combination, appear to have the potential to reduce the SCDF by 1E-05/rx-yr or the SLERF by 1E-06/rx-yr in one or both of the units:

- SEIS_LOSP — Loss of Offsite Power
- SEIS_0-30-5 — Relay Chatter - 480V/6.9kV SD BD GE HEA Breakers
- SEIS_1A-5 — 480V RMOV Board
- SEIS_23-5 — (NSSS Reactor Pressure Vessel
- SEIS_HINST — Seismically induced failure of HRA Main Control Room Instruments
- SEIS_0-16-2 — Instrument Line Very Small LOCA
- SEIS_3-4-1 — 120V AC Vital Inverter in Aux. Sub 1
- SEIS_23-6A — Steam Generator Support Failure
- SEIS_4-3 — PHMS Power Transformer
- SEIS_0-30-1 — Relay Chatter - SSPS Cabinets (R-048, R-051)
- HAMARV — Handwheel Operation of the SG Atmospheric Relief Valves S/G 1&4
- OP-LOCKOUT_69KSDB_S — Operator reset of 6900V Shutdown Board lockout relays
- HACIV — Isolate RCP seal water, thermal barrier injection and return lines on an SBO
- HAESBODG1_S — Align 225kVA 480V Diesel Generators -seismic
- HINST — Seismically induced failure of HRA MCR instrumentation (Unit 1 only)
- OP-LOCKOUT_EDG_S — Operator reset of EDG start lockout relays (Unit 2 only)
- HAFR2 — Restore TDAFWP speed control following initiator and loss of air

Of the above fragility groups and operator actions, the highest contributor to SCDF and SLERF for Units 1 and 2 is seismically-induced loss of offsite power (SEIS_LOSP). The staff notes that the switchyard and grid are outside the plant boundary and improvements made are very unlikely to be cost beneficial; however, potential reliability improvements were reviewed for equipment and operator actions that are used for mitigating SEIS LOSP. Based on the submittal, the dominant sequences with SEIS_LOSP that contribute to the SCDF involve loss of the shutdown boards due to relay chatter and failure to recover them, and failure of long-term cooldown with the steam generators via AFW when makeup from the CST or ERCW is not successful. The staff evaluated potential modifications of mitigation equipment (e.g., SEIS_0-30-5) and operator actions (e.g., OP-LOCKOUT_69KSDB_S, HAMARV, HAESBODG1_S) with the potential to reduce the SCDF by 1E-05/rx-yr or the SLERF by 1E-06/rx-yr in one or both units as discussed below.

The NRC staff notes that the cost of eliminating or significantly reducing risk from failure of the steam generator (SG) supports or failure of the reactor pressure vessel (RPV) supports is unlikely to be cost beneficial since the SG and RPV supports are major pieces of equipment and would constitute a large and disruptive modification. Failure of the RPV or SG supports are assumed to lead directly to core damage with no potential to mitigate the event.

In its submittal, the licensee provided the results of a sensitivity analysis (Case #25) that showed the composite reduction in SCDF and SLERF from increasing the fragilities of nuclear steam supply system (NSSS) components, including the reactor pressure vessel was less than 1.0E-05/rx-yr and 1.0E-06/rx-yr, respectively. Another sensitivity analysis (Case #23) showed the composite reduction in SCDF and SLERF from increasing the fragilities of selected components including the steam generator was less than 1.0E-05/rx-yr and 1.0E-06/rx-yr, respectively. As a result, the NRC staff did not pursue potential improvements to SEIS_23-5, or SEIS_23-6A.

During the audit, the licensee described the approach used to identify potential plant modifications for the remaining fragility groups and operator actions listed above (i.e., other than SEIS_LOSP, SEIS_23-5, or SEIS_23-6A). The licensee explained why plant or operational modifications were impractical or not cost-justified. The staff understands the term impractical to mean that the plant modification would be extensive and the cost would far exceed the monetary benefit associated with the risk reduction. The licensee explained that it examined the fragility groups and operator errors identified in Section 5 of the SPRA submittal in Tables 5.4-3, 5.4-4, 5.4-5, and 5.4-6 for SCDF and Tables 5.5-3, 5.5-4, 5.5-5, and 5.5-6 for SLERF that appeared to have the potential to reduce the SCDF by 1E-05/rx-yr or reduce the SLERF by 1E-06/rx-yr if eliminated. The licensee explained that many of the 31 sensitivity studies presented in Table 5.7-1 of the SPRA submittal were performed to investigate the potential risk reduction associated with possible plant modifications that address important seismic failures (e.g., Case 4 on the Refueling Water Storage Tank (RWST) fragility, Cases 9 and 10 on the fragility of main control room instrumentation, Case 12 on the Emergency Diesel Generator (EDG) and 6.9kV shutdown board lockout relay fragility, Case 14 on block wall fragility, Cases 16 and 17 on Essential Raw Cooling Water (ERCW) piping fragility, Case 23 on an aggregation of fragility groups, Case 25 on Nuclear Steam Supply System component fragility, and Case 31 on specific fragility groups including SEIS_0-30-5.)

The sensitivity studies were performed by assuming a significantly higher seismic capacity value or assuming the failure can be eliminated for each evaluated fragility group or operator action. The results from most of the sensitivity studies indicate that a significant risk decrease would not be achieved. For failures in which significant risk reduction may be achieved, including those failures cited above, the licensee stated that it considered all practical plant and operational changes that could credibly eliminate or significantly reduce the risk associated with the seismic failure. The licensee concluded that no plant modifications with the potential to reduce SCDF by 1E-05/rx-yr or SLERF by 1E-06/rx-yr were identified that were practical or could be cost-justified based on the extent of the modification.

For the SEIS_0-30-5 fragility group, the licensee provided the results of an additional sensitivity study considering an increase in the capacity of the fragility group from 0.84g (used on the baseline SPRA) to 1.65g. The results of the sensitivity study show that the change in SCDF for both units to be significantly below 1E-05/rx-yr (i.e., 1.55/2.11E-06/rx-yr) and the change in SLERF to be below 1E-06/rx-yr (i.e., 7.53/7.79E-07/rx-yr). The licensee explained that because the relays represented by the SEIS_0-30-5 fragility group are mounted on boards, the capacity of the relays cannot exceed the capacity of the boards for Unit 1; and for Unit 2 the relay capacity cannot exceed the capacity of the nearby block wall. The licensee further discussed possible plant modifications, such as moving the relays to a more rigid wall mounted panel or changing the relay reset from manual to automatic. The licensee explained that moving the relays to a more rigid wall mounted panel was determined not to be an option because Appendix R fire protection restrictions would prohibit their relocation and because the number of relays and boards involved (i.e., for a total of 12 installations) makes the modification impractical. The licensee also explained that changing the relay reset from manual to automatic could defeat the function of the lockout to protect the equipment. The licensee therefore concluded that the potential modifications associated with the SEIS_0-30-5 fragility group either did not provide substantial safety improvement or were impractical from an operational perspective.

For the HAMARV operator error, the licensee explained that this error pertains to local manual operator control of the steam generator (SG) atmospheric relief valves (ARVs) that fail closed on a loss of control air and cannot be controlled from the main control room. The licensee explained that the procedures associated with this operator action have been reviewed and that no procedure changes were identified that would significantly affect the human error probability. The licensee also explained that there were no practical plant modifications that would enable recovery of this failure from the main control room.

For lockout events (i.e., OP-LOCKOUT_69KSDB_S and OP-LOCKOUT_EDG_S) the licensee explained that resetting the lockout relays involves investigation and diagnosis by an operator and that these events cannot be replaced with an automatic action without the risk of equipment damage. The licensee also explained that the procedures associated with these operator actions have been reviewed and there have been no additional procedure changes identified that would significantly affect the human error probability.

For the HACIV operator error, the licensee explained that this error pertains to operator action to isolate the Reactor Cooling Pump (RCP) seal water, and thermal injection and return lines following a station blackout. The licensee explained that this action could only be performed as a local manual action at the isolation valves. The licensee also explained that automatic action was undesirable from an operational perspective and there were no practical plant modifications that would enable recovery of this failure from the main control room. The licensee also explained that the procedures associated with these operator action have been reviewed and there have been no procedure changes identified that would significantly affect the human error probability.

For the HAESBODG1_S operator error, the licensee explained that this error pertains to manual alignment of the 225kVA 480V FLEX diesel generators (DGs) to the 125 VDC Vital Battery Chargers, and that this function could only be automated if the FLEX DGs were redesigned and modified to be safety-related. The licensee also explained that the procedures associated with this operator action has been reviewed and there have been no additional procedure changes identified that would significantly affect the human error probability.

For the HAFR2 operator error, the licensee explained that this error pertains to operator action to control the speed of the turbine driven auxiliary feed pump when there is a failure of the [SG] level control valves. Accordingly, the action is manual recovery of the failure of a function that is automatically controlled and therefore, the recovery action itself cannot be automated. The licensee also explained that the procedures associated with this operator action have been reviewed and there have been no additional procedure changes identified that would significantly affect the human error probability.

For the HINST operator error (which encompasses the SEIS_HINST fragility group), the licensee explained that Sensitivity Study Case #10, as reported in the SPRA submittal, shows that halving the failure probability of control room instrumentation results in a decrease of less than 2.4 percent in the SCDF and SLERF. The licensee also stated that the panels were qualified by seismic testing. The licensee concluded that the capacity of the panels would have to be increased "significantly" beyond the assumption used in the sensitivity case (which is already conservative) in order to achieve a significant decrease in SCDF or SLERF for either unit.

In summary, for the operator actions that appeared to have potential cost-justified plant modifications, the licensee provided an explanation that there are no practical plant improvements that provide substantial safety improvement for Units 1 and 2 because (1) the operator action cannot be eliminated and replaced with an automatic action, (2) the operator action is relatively simple (HEP cannot be significantly improved), (3) no procedural improvements were identified that would significantly affect the HEP, and/or (4) significant re-design would be needed. The NRC staff reviewed each of these explanations and agreed that the conclusions are reasonable. As a result, the NRC staff did not pursue potential improvements to these operator actions or SSCs associated with the actions.

For other fragility group failures, (i.e., SEIS_1A-5, SEIS_0-16-2, SEIS_3-4-1, SEIS_4-3, and SEIS_0-30-1) the potential to reduce the SCDF by 1E-05/rx-yr or the SLERF by 1E-06/rx-yr appears only in combination with additional failures. The NRC staff considered these combinations of basic events in accordance with the SPRA Screening Guidance. Each of these combinations included one of the failures discussed above and potential modifications for those failures were determined to either not provide substantial safety improvement or were impractical. Further, it is not the intent of that aspect of the guidance to aggregate several disparate basic events that individually have RRW values close to the threshold. Therefore, potential improvements involving combinations were not pursued further.

Additionally, the NRC staff considered insights from the individual plant examination of external events (IPEEE) and SAMA analyses previously completed for Sequoyah to understand previous work done to identify substantial safety improvements and to further inform this review. NRC staff also considered the insights from "State-of-the-Art Reactor Consequence Analyses (SOARCA) Project: Sequoyah Integrated Deterministic and Uncertainty Analyses (NUREG/CR-7245)" (ADAMS Accession No. ML19296B786). No further potential improvements were found based on these previous evaluations. The Sequoyah SOARCA project highlighted the successful use of hydrogen igniters as effective in averting early containment failure. The SLERF accident sequences include failure of hydrogen igniters (igniters are stated to be seismically rugged, but may fail due to loss of their power transformers, loss of the shutdown board, or relay chatter), which is consistent with the Sequoyah SOARCA insights (i.e., the failure of the igniters results in early containment failure sequences).

Based on the analysis and information described above, the NRC staff concludes that no modifications are warranted in accordance with Title 10 of the *Code of Federal Regulations* (10 CFR) Section 50.109 to reduce SCDF and SLERF because no potential cost-justified substantial safety improvements were identified.

In accordance with Section 3.3.2 of NUREG/BR-0058, Revision 4, the NRC staff further evaluated Sequoyah accident sequences impacting the conditional probability of early containment failure or bypass (CPCFB) for seismic events to determine if any substantial safety improvements would reduce the SCDF and related SLERF of those sequences. All the dominant failures are already evaluated, as described above.

Based on the available information and engineering judgement, the NRC staff concluded that there were no further potential improvements to containment performance that would rise to the level of a substantial safety improvement or would warrant further regulatory analysis.

Conclusion

Based on the analysis of the submittal and supplemental information, the NRC staff concludes that no modifications are warranted under 10 CFR 50.109 because:

- the staff did not identify a potential modification necessary for adequate protection or compliance with existing regulations;
- no cost-justified substantial safety improvement was identified based on the estimated achievable reduction in SCDF and/or SLERF; and
- additional consideration of containment performance, as described in NUREG/BR-0058 and assessed via SLERF, did not identify a modification that would result in a substantial safety improvement.

Table 1: Importance Analysis Results of Top Contributors to SCDF for Unit 1

Fragility Group/Event	Descriptions	Failure Mode	RRW	MCR (/rx-yr)
Seismically Induced SSC Failures				
SEIS_LOSP	LOSP (Loss of Offsite Power)	Ceramic Insulators	2.481	7.58E-06
SEIS_0-30-5	Relay Chatter - 480V/6.9kV SD BD GE HEA Breakers	Functionality	1.618	4.85E-06
SEIS_23-5	NSSS Reactor Pressure Vessel	Anchorage	1.085	9.91E-07
SEIS_0-16-2	Instrument Line Very Small LOCA	Anchorage	1.073	8.64E-07
SEIS_VSLOCA	Seismic Very Small LOCA	Anchorage	1.040	4.83E-07
SEIS_3-4-1	120V AC Vital Inverter in Aux. sub 1	Anchorage	1.040	4.83E-07
SEIS_23-6	NSSS Steam Generator	Anchorage	1.025	3.05E-07
SEIS_23-6A	Steam Generator Support Failure	Anchorage	1.019	2.41E-07
SEIS_23-4	NSSS Pressurizer	Anchorage	1.016	2.03E-07
SEIS_HINST	Seismically induced failure of HRA Main Control Room (MCR) Instruments	Functionality	1.016	2.03E-07
SEIS_19-1	Condensate Storage Tank (CST)	Anchorage	1.014	1.78E-07
SEIS_1A-4	480V DG Aux. Board	Block Wall	1.013	1.65E-07
SEIS_19-9	Residual Heat Removal (RHR) Heat Exchanger	Anchorage	1.013	1.65E-07
SEIS_0-30-11	Relay Chatter - DGB Control Panel Kraus & Naimer relays	Functionality	1.012	1.52E-07
SEIS_3-4-2	120V AC Vital Inverter in Aux. sub 2	Anchorage	1.011	1.40E-07
SEIS_0-12	Block Walls in Diesel Generator Building	Block Wall	1.010	1.27E-07
SEIS_0-30-6	Relay Chatter - 480V/6.9kV SD BD, Pumps, Misc. Breakers	Functionality	1.009	1.14E-07
SEIS_0-1	Buried ERCW piping	Soil (Buried Piping) Failure	1.008	1.02E-07
SEIS_1C-1-2	U2 6.9kV Shutdown Board	Functionality	1.007	8.89E-08
SEIS_0-30-10	Relay Chatter - DGB Control Panel Square D Relays	Functionality	1.007	9.14E-08
Operator Errors				
OP-LOCKOUT_69KSDB_S	Operator reset of 6900V Shutdown Board lockout relays (seismic)	NA	1.634	4.93E-06
HAMARV	Handwheel Operation of the SG Atmospheric Relief Valves S/G 1&4	NA	1.531	4.41E-06
HAESBODG1_S	Align 225kVA 480V Diesel Generators (seismic)	NA	1.061	7.29E-07
HACSMU	Dependent HEP for NSF1ASWDHE-S4,RRCVENTDHE-S4	NA	1.021	2.55E-07
HASP1	Locally operate TD AFW pump after battery depletion	NA	1.019	2.31E-07
OP-LOCKOUT_EDG_S	Operator reset of EDG start lockout relays (seismic)	NA	1.018	2.29E-07
HINST	Seismically induced failure of HRA MCR instrumentation	NA	1.016	1.99E-07
HAFR2	Restore TDAFWP speed control following initiator and loss of air	NA	1.015	1.91E-07
HAAF2	Align ERCW supply to AFW pumps	NA	1.015	1.82E-07
HAOB3	Re-establish AFW Cooling following Loss of Unit 1 Vital Inst Power Board	NA	1.014	1.73E-07
HAESBO3MW_S	Align 6.9kV Diesel Generators (seismic)	NA	1.013	1.61E-07
HARR1	Align high-pressure recirculation, given auto swap over works	NA	1.006	7.62E-08
AFWOP1	Depressurize/cooldown to low-pressure injection following MLOCA	NA	1.005	6.35E-08

Table 2 Importance Analysis Results of Top Contributors to SLERF for Unit 1

Fragility Group/Event	Description	Failure Mode	RRW	MLR (/rx-yr)
Seismically Induced SSC Failures				
SEIS_LOSP	LOSP (Loss of Offsite Power)	Ceramic Insulators	1.658	2.73E-06
SEIS_0-30-5	Relay Chatter - 480V/6.9kV SD BD GE HEA Breakers	Functionality	1.319	1.66E-06
SEIS_1A-5	480V RMOV Board	Block wall	1.153	9.13E-07
SEIS_23-5	NSSS Reactor Pressure Vessel	Anchorage	1.134	8.10E-07
SEIS_HINST	Seismically induced failure of HRA Main Control Room Instruments	Functionality	1.103	6.38E-07
SEIS_0-16-2	Instrument Line Very Small LOCA	Anchorage	1.083	5.29E-07
SEIS_3-4-1	120V AC Vital Inverter in Aux. Sub 1	Anchorage	1.065	4.19E-07
SEIS_23-6A	Steam Generator Support Failure	Anchorage	1.043	2.81E-07
SEIS_4-3	PHMS Power Transformer	Anchorage	1.042	2.75E-07
SEIS_0-30-1	Relay Chatter - SSPS Cabinets (R-048, R-051)	Functionality	1.033	2.20E-07
SEIS_19-8	EDG Starting Air Tank	Anchorage	1.018	1.24E-07
SEIS_0-30-6	Relay Chatter - 480V/6.9kV SD BD, Pumps, Misc. Breakers	Functionality	1.018	1.24E-07
SEIS_VSLOCA	Seismic Very Small LOCA	Anchorage	1.016	1.10E-07
SEIS_0-30-11	Relay Chatter - DGB Control Panel Kraus & Naimer relays	Functionality	1.008	5.49E-08
SEIS_0-1	Buried ERCW piping	Soil (Buried Piping) Failure	1.007	4.81E-08
SEIS_19-9	Residual Heat Removal (RHR) Heat Exchanger	Anchorage	1.006	4.12E-08
SEIS_19-5	Refueling Water Storage Tank (RWST)	Tank Fragility	1.006	4.12E-08
Operator Errors				
HAMARV	Handwheel Operation of the SG Atmospheric Relief Valves S/G 1&4	NA	1.408	1.99E-06
OP-LOCKOUT_69KSDB_S	Operator reset of 6900V Shutdown Board lockout relays (seismic)	NA	1.361	1.82E-06
HACIV	Isolate RCP seal water, thermal barrier injection and return lines on a SBO	NA	1.235	1.30E-06
HAESBODG1_S	Align 225kVA 480V DGs (seismic)	NA	1.225	1.26E-06
HINST	Seismically induced failure of HRA MCR instrumentation	NA	1.225	1.26E-06
HAESBO3MW_S	Align 6.9kV Diesel Generators (seismic)	NA	1.014	9.68E-08
OP-LOCKOUT_EDG_S	Operator reset of EDG start lockout relays (seismic)	NA	1.012	8.17E-08
HART1	Manually trip reactor, given SSPS fails	NA	1.007	4.74E-08
AFWOP1	Depressurize/cooldown to low-pressure injection following MLOCA	NA	1.006	4.39E-08
HAAF2	Align ERCW supply to AFW pumps	NA	1.005	3.57E-08
HAOB3	Re-establish AFW Cooling following Loss of Unit 1 Vital Inst Power Board	NA	1.005	3.43E-08

Table 3 Importance Analysis Results of Top Contributors to SCDF for Unit 2

Fragility Group/Event	Description	Failure Mode	RRW	MCR (/rx-yr)
Seismically Induced SSC Failures				
SEIS_LOSP	LOSP (Loss of Offsite Power)	Ceramic Insulators	3.003	1.00E-05
SEIS_0-30-5	Relay Chatter - 480V/6.9kV SD BD GE HEA Breakers	Functionality	1.919	7.19E-06
SEIS_3-4-1	120V AC Vital Inverter in Aux. sub 1	Anchorage	1.093	1.28E-06
SEIS_0-16-2	Instrument Line Very Small LOCA	Anchorage	1.082	1.14E-06
SEIS_23-5	NSSS Reactor Pressure Vessel	Anchorage	1.063	8.85E-07
SEIS_VSLOCA	Seismic Very Small LOCA	Anchorage	1.042	6.00E-07
SEIS_3-4-2	120V AC Vital Inverter in Aux. sub 2	Anchorage	1.040	5.70E-07
SEIS_23-6	NSSS Steam Generator	Anchorage	1.025	3.60E-07
SEIS_0-30-11	Relay Chatter - DGB Control Panel Kraus & Naimer relays	Functionality	1.021	3.15E-07
SEIS_23-6A	Steam Generator Support Failure	Anchorage	1.016	2.40E-07
SEIS_HINST	Seismically induced failure of MCR Instruments	Functionality	1.015	2.25E-07
SEIS_23-4	NSSSS Pressurizer	Anchorage	1.014	2.10E-07
SEIS_0-30-6	Relay Chatter - 480V/6.9kV SD BD, Pumps, Misc. Breakers	Functionality	1.014	2.10E-07
SEIS_1A-4	480V DG Aux. Board	Block Wall	1.013	1.95E-07
SEIS_0-30-10	Relay Chatter - DGB Cntrl Pnl Square D Relays	Functionality	1.012	1.80E-07
SEIS_19-9	Residual Heat Removal (RHR) Heat Exchanger	Anchorage	1.012	1.80E-07
SEIS_0-12	Block Walls in Diesel Generator Building	Block Wall	1.011	1.65E-07
SEIS_0-1	Buried ERCW piping	Soil (Buried Piping) Fails	1.009	1.35E-07
SEIS_FLEX_BUS480	FLEX 480V DG Bus Panel	Functionality	1.008	1.20E-07
SEIS_16-1	Aux. Control Air Compressor	Functionality	1.008	1.20E-07
SEIS_19-1	Condensate Storage Tank (CST)	Anchorage	1.008	1.20E-07
SEIS_0-30-9	Aux. Control Air Compressor	Functionality	1.007	1.05E-07
SEIS_1B-1	480V Shutdown Board	Anchorage	1.005	7.50E-08
SEIS_12-2	ERCW Pump	Functionality	1.005	7.50E-08
SEIS_IF	Seismic Induced Flood	Block Wall	1.005	7.50E-08
Operator Errors				
OP-LOCKOUT_69KSDB_S	Operator reset of 6900V Shutdown Board lockout relays (seismic)	NA	1.931	7.23E-06
HAMARV	Handwheel Operation of the SG Atmospheric Relief Valves S/G 1&4	NA	1.786	6.60E-06
HAESBODG1_S	Align 225kVA 480V DGs (seismic)	NA	1.030	4.32E-07
OP-LOCKOUT_EDG_S	Operator reset of EDG start lockout relays (seismic)	NA	1.034	4.89E-07
HAFR2	Restore TDAFWP speed control following initiator and loss of air	NA	1.025	3.69E-07
HAESBO3MW_S	Align 6.9kV DGss (seismic)	NA	1.020	2.97E-07
HASP1	Locally operate TD AFW pump after battery depletion	NA	1.019	2.78E-07

Fragility Group/Event	Description	Failure Mode	RRW	MCR (/rx-yr)
HACSMU	Dependent HEP for NSF1ASWDHE-S4,RRCVENTDHE-S4	NA	1.016	2.34E-07
HINST	Seismically induced failure of HRA MCR instrumentation	NA	1.015	2.27E-07
HAAF2	Align ERCW supply to AFW pumps	NA	1.013	1.95E-07
AFWOP1	Depressurize/cooldown to low-pressure injection following MLOCA	NA	1.001	1.49E-08
OPLOCKOUT_480VSDBD_S	Operator reset of 480V Shutdown Board lockout relays (seismic)	NA	1.001	1.08E-08
HARR1	Align high-pressure recirculation, given auto swap over works	NA	1.001	8.25E-09

Table 4 Importance Analysis Results of Top Fragility Group Contributors to SLERF for Unit 2

Fragility Group/Event	Description	Failure Mode	RRW	MLR (/rx-yr)
	Seismically Induced SSC Failures			
SEIS_LOSP	LOSP (Loss of Offsite Power)	Ceramic Insulators	1.704	2.86E-06
SEIS_0-30-5	Relay Chatter - 480V/6.9kV SD BD GE HEA Breakers	Functionality	1.406	2.00E-06
SEIS_1A-5	480V RMOV Board	Block wall	1.119	7.35E-07
SEIS_23-5	NSSS Reactor Pressure Vessel	Anchorage	1.115	7.14E-07
SEIS_HINST	Seismically induced failure of HRA Main Control Room (MCR) Instruments	Functionality	1.100	6.31E-07
SEIS_3-4-1	120V AC Vital Inverter in Aux. sub 1	Anchorage	1.070	4.50E-07
SEIS_0-16-2	Instrument Line Very Small LOCA	Anchorage	1.063	4.09E-07
SEIS_0-30-1	Relay Chatter - SSPS Cabinets (R-048, R-051)	Functionality	1.047	3.12E-07
SEIS_23-6A	Steam Generator Support Failure	Anchorage	1.045	2.98E-07
SEIS_3-4-2	120V AC Vital Inverter in Aux. sub 2	Anchorage	1.034	2.29E-07
SEIS_4-3	PHMS Power Transformer	Anchorage	1.034	2.29E-07
SEIS_VSLOCA	Seismic Very Small LOCA	Anchorage	1.024	1.59E-07
SEIS_0-30-6	Relay Chatter - 480V/6.9kV SD BD, Pumps, Misc. Breakers	Functionality	1.013	9.01E-08
SEIS_19-8	EDG Starting Air Tank	Anchorage	1.010	6.93E-08
SEIS_19-9	Residual Heat Removal (RHR) Heat Exchanger	Anchorage	1.006	4.16E-08
SEIS_19-5	Refueling Water Storage Tank (RWST)	Tank Fragility	1.006	4.16E-08
SEIS_0-30-11	Relay Chatter - DGB Control Panel Kraus & Naimer relays	Functionality	1.005	3.47E-08
	Operator Errors			
OP-LOCKOUT_69KSDB_S	Operator reset of 6900V Shutdown Board lockout relays (seismic)	NA	1.433	2.09E-06
HAMARV	Handwheel Operation of the SG Atmospheric Relief Valves S/G 1&4	NA	1.404	2.00E-06
HAESBODG1_S	Align 225kVA 480V DGs (seismic)	NA	1.340	1.76E-06
HACIV	Isolate RCP seal water, thermal barrier injection and return lines on a SBO	NA	1.176	1.04E-06
HINST	Seismically induced failure of HRA MCR instrumentation	NA	1.100	6.31E-07
HAPRZ	Depressurization of the RCS using pressurizer PORVs (Level II only)	NA	1.020	1.33E-07
HAESBO3MW_S	Align 6.9kV Diesel Generators (seismic)	NA	1.008	5.54E-08
OP-LOCKOUT_EDG_S	Operator reset of EDG start lockout relays (seismic)	NA	1.008	5.47E-08
HASP1	Locally operate TD AFW pump after battery depletion	NA	1.006	4.09E-08
HAFR2	Restore TDAFWP speed control following initiator and loss of air	NA	1.006	4.02E-08
HAAF2	Align ERCW supply to AFW pumps	NA	1.005	3.74E-08

AUDIT SUMMARY BY THE OFFICE OF NUCLEAR REACTOR REGULATION RELATED TO

SEQUOYAH NUCLEAR PLANT, UNITS 1 AND 2

SUBMITTAL OF SEISMIC PROBABILISTIC RISK ASSESSMENT ASSOCIATED WITH

REEVALUATED SEISMIC HAZARD IMPLEMENTATION OF THE

NEAR-TERM TASK FORCE RECOMMENDATION 2.1: SEISMIC

(EPID NO. L-2019-JLD-0013)

BACKGROUND AND AUDIT BASIS

By letter dated March 12, 2012 (Agencywide Documents Access and Management System (ADAMS) Accession No. ML12053A340), the U.S. Nuclear Regulatory Commission (NRC) issued a request for information pursuant to Title 10 of the *Code of Federal Regulations* (10 CFR), Section 50.54(f) (hereafter referred to as the 50.54(f) letter). Enclosure 1 to the 50.54(f) letter requested that licensees reevaluate the seismic hazards for their sites using present-day methods and regulatory guidance used by the NRC staff when reviewing applications for early site permits and combined licenses.

By letter dated October 27, 2015 (ADAMS Accession No. ML15194A015), the NRC made a determination of which licensees were to perform: (1) a Seismic Probabilistic Risk Assessment (SPRA), (2) limited scope evaluations, or (3) no further actions based on a comparison of the reevaluated seismic hazard and the site's design-basis earthquake. (Note: Some plant-specific changes regarding whether an SPRA was needed or limited scope evaluations were needed at certain sites have occurred since the issuance of the October 27, 2015, letter).

By letter dated July 6, 2017 (ADAMS Accession No. ML17177A446), the NRC issued a generic audit plan and entered into the audit process described in Office Instruction LIC-111, "Regulatory Audits," dated December 29, 2008 (ADAMS Accession No. ML082900195), to assist in the timely and efficient closure of activities associated with the letter issued pursuant to Title 10 CFR Part 50, Section 50.54(f). The list of applicable licensees in Enclosure 1 to the July 6, 2017, letter included Tennessee Valley Authority (TVA) as the licensee for Sequoyah Nuclear Plant, Units 1 and 2 (Sequoyah).

REGULATORY AUDIT SCOPE AND METHODOLOGY

The areas of focus for the regulatory audit are the information contained in the SPRA submittal and all associated and relevant supporting documentation used in the development of the SPRA submittal including, but not limited to, methodology, process information, calculations, computer models, etc.

AUDIT ACTIVITIES

The NRC staff developed questions to verify information in the licensee's submittal and to gain understanding of non-docketed information that supports the docketed SPRA report. The staff's request for supporting documents and clarification questions dated November 19, 2019, February 19, 2020, and April 9, 2020 (ADAMS Accession Nos. ML19323E793, ML20056D696, and ML20161A390, respectively), were sent to the licensee to support the audit.

The licensee provided clarifying information in the following areas:

- Seismically induced soil settlement in the fragility analyses for the Diesel Generator Building (DGB) and the Additional Diesel Generator Building (ADGB)
- Screening approach used for selection of SSCs for seismic fragility analysis
- Scaling of Watts Bar NSSS component fragilities to Sequoyah NSSS components
- Scaling of Watts Bar structural fragilities to identify Sequoyah fragilities
- Evaluation of potential plant modifications that could reduce SCDF and SLERF (none identified).

The licensee's response to the questions aided in the staff's understanding of the Sequoyah SPRA docketed submittal. Following the review of the licensee's response and the supporting documents provided by the licensee on the eportal, the staff determined that no additional documentation or information was needed to supplement the docketed SPRA submittal.

<u>DOCUMENTS AUDITED</u>

- Jensen-Hughes Report 006063-RPT-01, Revision 0, "Sequoyah SPRA F&O Independent Assessment & Focused-Scope-Peer Review," April 2019.
- ENERCON Report TVAESQN010-REPT-002, Revision 2, "Sequoyah Units 1 & 2 Seismic PRA Chatter Analysis Report," July 2018.
- SC Solutions Report SQN-17-001, Revision 2, "Sequoyah SPRA – Structural Response Analysis Report," November 2018.
- ENERCON Report TVAESQN010-REPT-001, Revision 1, "Units 1 & 2 Sequoyah Nuclear Power Plant Seismic Probabilistic Risk Assessment – Fragility Analysis Notebook," February 2019.
- Report CJC-SQN-C-001, Revision 1, "Updated Soil Failure and Fragility Analysis for the Sequoyah Nuclear Plant (SQN)," April 2019.
- PWROG-18002-P, Revision 0, "Peer Review of the Sequoyah Seismic Probabilistic Risk Assessment," Westinghouse PWR Owners Group, Risk Management Committee, July 2018
- TVA Calculation MDN0009992017000045, Revision 1, "SQN Seismic PRA Quantification, Sensitivity and Uncertainty Notebook," June 11, 2019.
- TVA Calculation MDN000NA2017000042, Revision 1, "SQN Seismic PRA Seismic-Fire Interaction," March 22, 2019.

- TVA Calculation MDN00000020100203, Revision 3, “SQN Internal Flooding,” August 18, 2014.
- TVA PRA Evaluation SQN-0-17-095, Revision 0, “The Selection of Seismic Flooding Scenarios,” September 7, 2017.
- TVA Calculation MDN0009992017000043, Revision 1, “SQN Probabilistic Risk Assessment - Seismic PRA Human Reliability Analysis,” January 23, 2019.
- TVA Calculation MDN00000020100200, Revision 3, “SQN Probabilistic Risk Assessment – Summary Notebook,” August 4, 2014.
- TVA Calculation MDN0009992017000044, Revision 2, “SQN Probabilistic Risk Assessment – SQN Seismic PRA Methodology, Inputs, and Model Notebook,” October 15, 2019.

OPEN ITEMS AND REQUEST FOR INFORMATION

There were no open items identified by the NRC staff that required proposed closure paths and there were no requests for information discussed or planned to be issued based on the audit.

DEVIATIONS FROM AUDIT PLAN

There were no deviations from the generic audit plan dated July 6, 2017.

AUDIT CONCLUSION

The issuance of this document, containing the staff’s review of the SPRA submittal, concludes the SPRA audit process for Sequoyah.

SUBJECT: SEQUOYAH NUCLEAR PLANT, UNITS 1 AND 2 – STAFF REVIEW OF SEISMIC PROBABILISTIC RISK ASSESSMENT ASSOCIATED WITH REEVALUATED SEISMIC HAZARD IMPLEMENTATION OF THE NEAR-TERM TASK FORCE RECOMMENDATION 2.1: SEISMIC DATED JULY 17, 2020

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OFFICE	NRR/DORL/LPMB/PM*	NRR/DANU/UARL/LA*	NRR/DORL/LPMB/BC*
NAME	SPhilpott	SLent	DWrona
DATE	6/17/2020	6/12/2020	6/18/2020
OFFICE	NRR/DRA/D*	NRR/DEX/DD*	NRR/DORL/DD*
NAME	MFranovich	JMarshall	MShams
DATE	7/17/2020	7/9/2020	07/17/2020

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