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

September 18, 2019

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

SUBJECT: SEQUOYAH NUCLEAR PLANT, UNITS 1 AND 2 - ISSUANCE OF AMENDMENT NOS. 346 AND 340 RE: REQUEST TO ADOPT 10 CFR 50.69, "RISK-INFORMED CATEGORIZATION AND TREATMENT OF STRUCTURES, SYSTEMS, AND COMPONENTS FOR NUCLEAR POWER REACTORS" (EPID L-2018-LLA-0066)

Dear Mr. Barstow:

The U.S. Nuclear Regulatory Commission (the Commission) has issued the enclosed Amendment Nos. 346 and 340 to Renewed Facility Operating License Nos. DPR-77 and DPR-79 for the Sequoyah Nuclear Plant, Units 1 and 2, respectively. The amendments consist of changes to the Renewed Facility Operating Licenses in response to your application dated March 16, 2018, as supplemented by letter dated March 21, 2019.

The amendments add a new license condition to the Renewed Facility Operating Licenses to allow the implementation of Title 10 of the *Code of Federal Regulations* (10 CFR) Section 50.69, "Risk-informed categorization and treatment of structures, systems, and components for nuclear power reactors." The provisions of 10 CFR 50.69 allow adjustment of the scope of structures, systems, and components subject to special treatment requirements (e.g., quality assurance, testing, inspection, condition monitoring, assessment, and evaluation) based on an integrated and systematic risk-informed process that includes several approaches and methods for categorizing structures, systems, and components according to their safety significance.

A copy of the related Safety Evaluation is also enclosed. A Notice of Issuance will be included in the Commission's biweekly *Federal Register* notice.

Sincerely,

Andrew Hon, Project Manager
Plant Licensing Branch II-2
Division of Operating Reactor Licensing
Office of Nuclear Reactor Regulation

Docket Nos. 50-327 and 50-328

Enclosures:
1. Amendment No. 346 to DPR-77
2. Amendment No. 340 to DPR-79
3. Safety Evaluation

cc: Listserv

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

TENNESSEE VALLEY AUTHORITY

DOCKET NO. 50-327

SEQUOYAH NUCLEAR PLANT, UNIT 1

AMENDMENT TO RENEWED FACILITY OPERATING LICENSE

Amendment No. 346
Renewed License No. DPR-77

1. The Nuclear Regulatory Commission (the Commission) has found that:

   A. The application for amendment by the Tennessee Valley Authority (the licensee), dated March 16, 2018, as supplemented by letter dated March 21, 2019, complies with the standards and requirements of the Atomic Energy Act of 1954, as amended (the Act), and the Commission's rules and regulations set forth in 10 CFR Chapter I;

   B. The facility will operate in conformity with the application, as amended; the provisions of the Act; and the rules and regulations of the Commission;

   C. There is reasonable assurance (i) that the activities authorized by this amendment can be conducted without endangering the health and safety of the public, and (ii) that such activities will be conducted in compliance with the Commission's regulations;

   D. The issuance of this amendment will not be inimical to the common defense and security or to the health and safety of the public; and

   E. The issuance of this amendment is in accordance with 10 CFR Part 51 of the Commission's regulations and all applicable requirements have been satisfied.

2. Accordingly, the license is amended by changes as indicated in the attachment to this license amendment, and Renewed Facility Operating License No. DPR-77 is hereby amended to add paragraph (33) to read as follows:

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

(1) TVA is approved to implement 10 CFR 50.69 using the processes for categorization of Risk Informed Safety Class (RISC)-1, RISC-2, RISC-3, and RISC-4 structures, systems, and components (SSCs) using: Probabilistic Risk Assessment (PRA) model to evaluate risk associated with internal events, including internal flooding; using the fire safe shutdown equipment list in the SQN Fire Protection Report referenced in the Updated Final Safety Analysis Report to evaluate internal fire events; the NUMARC 91-06 shutdown safety assessment process to assess shutdown risk; the Arkansas Nuclear One, Unit 2 (ANO-2) passive categorization method to assess passive component risk for Class 2 and Class 3 SSCs and their associated supports; and the results of non PRA evaluations that are based on the IPEEE Screening Assessment for External Hazards, i.e., seismic margin analysis (SMA) to evaluate seismic risk, and a screening of other external hazards updated using the criteria in the endorsed ASME/ANS RA-Sa-2009 PRA Standard for other external hazard screening significance; as specified in Unit 1 License Amendment 346.

(2) Prior to implementation of the provisions of 10CFR 50.69, TVA shall complete the items below;

a. Items listed in Enclosure 1, Attachment 1, "SQN 10 CFR 50.69 PRA Implementation Items," in TVA letter CNL-19-002, "Response to Request for Additional Information Regarding Application to Modify Sequoyah Nuclear Plant Units 1 and 2, Application to Adopt 10 CFR 50.69, "Risk-informed Categorization and Treatment of Structures, Systems, and Components for Nuclear Power Reactors, (SQN-TS-17-06) (EPID: L-2018-LLA-0066)," dated March 21, 2019.

(3) Prior NRC approval, under 10 CFR 50.90, is required for a change to the categorization process specified above (e.g., change from a seismic margins approach to a seismic probabilistic risk assessment approach, change from alternative method for internal fire to a fire probabilistic risk assessment approach).

3. This license amendment is effective as of the date of its issuance and shall be implemented no later than 60 days from the date of its issuance.

FOR THE NUCLEAR REGULATORY COMMISSION

Ed Miller FOR US

Undine Shoop, Chief
Plant Licensing Branch II-2
Division of Operating Reactor Licensing
Office of Nuclear Reactor Regulation

Attachment:
Changes to the Renewed
Facility Operating License

Date of Issuance: September 18, 2019

ATTACHMENT TO LICENSE AMENDMENT NO. 346

SEQUOYAH NUCLEAR PLANT, UNIT 1

RENEWED FACILITY OPERATING LICENSE NO. DPR-77

DOCKET NO. 50-327

Replace the following page of the Renewed Facility Operating License with the attached revised pages. The revised pages are identified by amendment number and contain marginal lines indicating the areas of change.

Remove	Insert
14a	14a
--	14b

(d) For SRs that existed prior to this amendment that have modified acceptance criteria, the first performance subject to the modified acceptance criteria is due at the end of the first Surveillance interval that began on the date the Surveillance was last performed prior to the implementation of this amendment.

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

(1) TVA is approved to implement 10 CFR 50.69 using the processes for categorization of Risk Informed Safety Class (RISC)-1, RISC-2, RISC-3, and RISC-4 structures, systems, and components (SSCs) using: Probabilistic Risk Assessment (PRA) model to evaluate risk associated with internal events, including internal flooding; using the fire safe shutdown equipment list in the SQN Fire Protection Report referenced in the Updated Final Safety Analysis Report to evaluate internal fire events; the NUMARC 96-01 shutdown safety assessment process to assess shutdown risk; the Arkansas Nuclear One, Unit 2 (ANO-2) passive categorization method to assess passive component risk for Class 2 and Class 3 SSCs and their associated supports; and the results of non PRA evaluations that are based on the IPEEE Screening Assessment for External Hazards, i.e., seismic margin analysis (SMA) to evaluate seismic risk, and a screening of other external hazards updated using the criteria in the endorsed ASME/ANS RA-Sa-2009 PRA Standard for other external hazard screening significance; as specified in Unit 1 License Amendment 346.

(2) Prior to implementation of the provisions of 10CFR 50.69, TVA shall complete the items below;

a. Items listed in Enclosure 1, Attachment 1, "SQN 10 CFR 50.69 PRA Implementation Items," in TVA letter CNL-19-002, "Response to Request for Additional Information Regarding Application to Modify Sequoyah Nuclear Plant Units 1 and 2, Application to Adopt 10 CFR 50.69, "Risk-informed Categorization and Treatment of Structures, Systems, and Components for Nuclear Power Reactors, (SQN-TS-17-06)(EPID: L-2018-LLA-0066)," dated March 21, 2019.

(3) Prior NRC approval, under 10 CFR 50.90, is required for a change to the categorization process specified above (e.g., change from a seismic margins approach to a seismic probabilistic risk assessment approach, change from alternative method for internal fire to a fire probabilistic risk assessment approach).

D. Exemptions from certain requirements of Appendices G and J to 10 CFR Part 50 are described in the Office of Nuclear Reactor Regulation's Safety Evaluation Report, Supplement No. 1. These exemptions are authorized by law and will not endanger life or property or the common defense and security and are otherwise in the public interest. The exemptions are, therefore, hereby granted. The granting of these exemptions are authorized with the issuance of the License for Fuel Loading and Low Power Testing, dated February 29, 1980. The facility will operate, to the extent authorized herein, Act, and the regulations of the Commission.

E. Physical Protection

(1) The licensee shall fully implement and maintain in effect all provisions of the Commission-approved physical security, training and qualification, and safeguards contingency plans including amendments made pursuant to provisions of the Miscellaneous Amendments and Search Requirements revisions to 10 CFR 73.55 (51 FR 27817 and 27822) and to the authority of 10 CFR 50.90 and 10 CFR 50.54(p). The combined set of plans, which contain Safeguards Information protected under 10 CFR 73.21, is entitled: "Sequoyah Nuclear Plant Security Plan, Training And Qualification Plan, And Safeguards Contingency Plan" submitted by letter dated May 8, 2006.

(2) The licensee shall fully implement and maintain in effect all provisions of the Commission-approved cyber security plan (CSP), including changes made pursuant to the authority of 10 CFR 50.90 and 10 CFR 50.54(p). The licensee CSP was approved by License Amendment No. 329, as amended by changes approved by License Amendment Nos. 333 and 337.

TENNESSEE VALLEY AUTHORITY

DOCKET NO. 50-328

SEQUOYAH NUCLEAR PLANT, UNIT 2

AMENDMENT TO FACILITY OPERATING LICENSE

Amendment No. 340
Renewed License No. DPR-79

1. The Nuclear Regulatory Commission (the Commission) has found that:

   A. The application for amendment by the Tennessee Valley Authority (the licensee), dated March 16, 2018, as supplemented by letter dated March 21, 2019, complies with the standards and requirements of the Atomic Energy Act of 1954, as amended (the Act), and the Commission's rules and regulations set forth in 10 CFR Chapter I;

   B. The facility will operate in conformity with the application, as amended; the provisions of the Act; and the rules and regulations of the Commission;

   C. There is reasonable assurance (i) that the activities authorized by this amendment can be conducted without endangering the health and safety of the public, and (ii) that such activities will be conducted in compliance with the Commission's regulations;

   D. The issuance of this amendment will not be inimical to the common defense and security or to the health and safety of the public; and

   E. The issuance of this amendment is in accordance with 10 CFR Part 51 of the Commission's regulations and all applicable requirements have been satisfied.

2. Accordingly, the license is amended by changes as indicated in the attachment to this license amendment, and Renewed Facility Operating License No. DPR-79 is hereby amended to add paragraph (26) to read as follows:

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

(1) TVA is approved to implement 10 CFR 50.69 using the processes for categorization of Risk Informed Safety Class (RISC)-1, RISC-2, RISC-3, and RISC-4 structures, systems, and components (SSCs) using: Probabilistic Risk Assessment (PRA) model to evaluate risk associated with internal events, including internal flooding; using the fire safe shutdown equipment list in the SQN Fire Protection Report referenced in the Updated Final Safety Analysis Report to evaluate internal fire events; the NUMARC 91-06 shutdown safety assessment process to assess shutdown risk; the Arkansas Nuclear One, Unit 2 (ANO-2) passive categorization method to assess passive component risk for Class 2 and Class 3 SSCs and their associated supports; and the results of non PRA evaluations that are based on the IPEEE Screening Assessment for External Hazards, i.e., seismic margin analysis (SMA) to evaluate seismic risk, and a screening of other external hazards updated using the criteria in the endorsed ASME/ANS RA-Sa-2009 PRA Standard for other external hazard screening significance; as specified in Unit 2 License Amendment 340.

(2) Prior to implementation of the provisions of 10CFR 50.69, TVA shall complete the items below;

a. Items listed in Enclosure 1, Attachment 1, "SQN 10 CFR 50.69 PRA Implementation Items," in TVA letter CNL-19-002, "Response to Request for Additional Information Regarding Application to Modify Sequoyah Nuclear Plant Units 1 and 2, Application to Adopt 10 CFR 50.69, "Risk-informed Categorization and Treatment of Structures, Systems, and Components for Nuclear Power Reactors, (SQN-TS-17-06) (EPID: L-2018-LLA-0066)," dated March 21, 2019.

(3) Prior NRC approval, under 10 CFR 50.90, is required for a change to the categorization process specified above (e.g., change from a seismic margins approach to a seismic probabilistic risk assessment approach, change from alternative method for internal fire to a fire probabilistic risk assessment approach).

3. This license amendment is effective as of the date of its issuance and shall be implemented no later than 60 days from the date of its issuance.

FOR THE NUCLEAR REGULATORY COMMISSION

[illegible] FOR US

Undine Shoop, Chief
Plant Licensing Branch II-2
Division of Operating Reactor Licensing
Office of Nuclear Reactor Regulation

Attachment:
Changes to the Renewed
Facility Operating License

Date of Issuance: September 18, 2019

ATTACHMENT TO LICENSE AMENDMENT NO. 340

SEQUOYAH NUCLEAR PLANT, UNIT 2

RENEWED FACILITY OPERATING LICENSE NO. DPR-79

DOCKET NO. 50-328

Replace the following page of the Renewed Facility Operating License with the attached revised pages. The revised pages are identified by amendment number and contain marginal lines indicating the areas of change.

Remove	Insert
13a	13a
--	13b

relocation of the requirements to the specified documents, as described in Table R, Relocated Specifications and Removed Detail Changes, attached to the NRC staff's Safety Evaluation, which is enclosed in this amendment.

2. <u>Schedule for New and Revised Surveillance Requirements (SRs) The schedule for performing SRs that are new or revised in License Amendment 327 shall be as follows:</u>

   (a) For SRs that are new in this amendment, the first performance is due at the end of the first Surveillance interval, which begins on the date of implementation of this amendment.

   (b) For SRs that existed prior to this amendment, whose intervals of performance are being reduced, the first reduced Surveillance interval begins upon completion of the first Surveillance performed after implementation of this amendment.

   (c) For SRs that existed prior to this amendment, whose intervals of performance are being extended, the first extended Surveillance interval begins upon completion of the last Surveillance performed prior to implementation of this amendment.

   (d) For SRs that existed prior to this amendment that have modified acceptance criteria, the first performance subject to the modified acceptance criteria is due at the end of the first Surveillance interval that began on the date the Surveillance was last performed prior to the implementation of this amendment.

(26) TVA will implement the compensatory measures described in Section 3.8, "Additional Compensatory Measures," of TVA letter CNL-19-072, dated July 14, 2019, during the timeframe the Upper Range Reactor Vessel Level Instrumentation is not required to be operable for the remainder of Cycle 23. If the Upper Range Reactor Vessel Level Instrumentation is returned to operable status prior to the end of Cycle 23, then these compensatory measures are no longer required.

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

   (1) TVA is approved to implement 10 CFR 50.69 using the processes for categorization of Risk Informed Safety Class (RISC)-1, RISC-2, RISC-3, and RISC-4 structures, systems, and components (SSCs) using: Probabilistic Risk Assessment (PRA) model to evaluate risk associated with internal events, including internal flooding; using the fire safe shutdown equipment list in the SQN Fire Protection Report referenced in the Updated Final Safety Analysis Report to evaluate internal fire events; the NUMARC 96-01 shutdown safety assessment process to assess shutdown risk; the Arkansas Nuclear One, Unit 2 (ANO-2) passive categorization method to assess passive component risk for Class 2 and Class 3 SSCs and their associated supports; and the results of non PRA evaluations that are based on the

IPEEE Screening Assessment for External Hazards, i.e., seismic margin analysis (SMA) to evaluate seismic risk, and a screening of other external hazards updated using the criteria in the endorsed ASME/ANS RA-Sa-2009 PRA Standard for other external hazard screening significance; as specified in Unit 2 License Amendment 340.

(2) Prior to implementation of the provisions of 10CFR 50.69, TVA shall complete the items below;

a. Items listed in Attachment 1, "SQN 10 CFR 50.69 PRA Implementation Items," in TVA letter CNL-19-002, "Response to Request for Additional Information Regarding Application to Modify Sequoyah Nuclear Plant Units 1 and 2, Application to Adopt 10 CFR 50.69, "Risk-informed Categorization and Treatment of Structures, Systems, and Components for Nuclear Power Reactors, (SQN-TS-17-06) (EPID: L-2018-LLA-0066)," dated March 21, 2019.

(3) Prior NRC approval, under 10 CFR 50.90, is required for a change to the categorization process specified above (e.g., change from a seismic margins approach to a seismic probabilistic risk assessment approach, change from alternative method for internal fire to a fire probabilistic risk assessment approach).

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

SAFETY EVALUATION BY THE OFFICE OF NUCLEAR REACTOR REGULATION

RELATED TO AMENDMENT NO. 346 TO RENEWED FACILITY

OPERATING LICENSE NO. DPR-77 AND AMENDMENT NO. 340 TO

RENEWED FACILITY OPERATING LICENSE NO. DPR-79

TENNESSEE VALLEY AUTHORITY

SEQUOYAH NUCLEAR PLANT, UNITS 1 AND 2

DOCKET NOS. 50-327 AND 50-328

1.0 INTRODUCTION

By letter dated March 16, 2018 (Reference 1), as supplemented by the letter dated March 21, 2019 (Reference 2), Tennessee Valley Authority (TVA, the licensee), submitted a license amendment request (LAR) for the Sequoyah Nuclear Plant (Sequoyah or SQN), Units 1 and 2. The licensee proposed to add a new license condition to the TVA Renewed Facility Operating Licenses (RFOLs) to allow the implementation of Title 10 of the *Code of Federal Regulations* (10 CFR) Section 50.69, "Risk-informed categorization and treatment of structures, systems, and components for nuclear power reactors." The provisions of 10 CFR 50.69 allow adjustment of the scope of structures, systems, and components (SSCs) subject to special treatment requirements (e.g., quality assurance, testing, inspection, condition monitoring, assessment, and evaluation) based on an integrated and systematic risk-informed process that includes several approaches[1] and methods for categorizing SSCs according to their safety significance.

By e-mail dated January 15, 2019 (Reference 3), the U.S. Nuclear Regulatory Commission (NRC or the Commission) staff requested additional information from the licensee. The licensee responded to the requests for additional information (RAIs) in the supplemental letter dated March 21, 2019. The supplemental letter provided additional information that clarified the application, did not expand the scope of the application as originally noticed, and did not change the NRC staff's proposed no significant hazards consideration determination as published in the *Federal Register* on August 28, 2018 (83 FR 43908).

[1] Regulatory Guide (RG) 1.201, Revision 1, "Guidelines for Categorizing Structures, Systems, and Components in Nuclear Power Plants According to Their Safety Significance," May 2006 (Reference 4), describes the SSC categorization process in its entirety as an overarching approach that includes multiple approaches and methods identified for a probabilistic risk assessment (PRA) hazard and non-PRA methods.

2.0 REGULATORY EVALUATION

2.1 Risk-Informed Categorization and Treatment of Structures, Systems, and Components

The risk-informed approach to regulation enhances and extends the traditional deterministic regulation by considering risk in a comprehensive manner. Specifically, a probabilistic approach allows consideration of a broader set of potential challenges to safety, providing a logical means for prioritizing these challenges based on safety significance, and allowing consideration of a broader set of resources to defend against these challenges. PRAs address credible initiating events by assessing the event frequency. Mitigating system reliability is then assessed, including the potential for common cause failures.

To take advantage of the safety enhancements available through the use of PRA, the NRC promulgated a new regulation, 10 CFR 50.69, in the *Federal Register* on November 22, 2004 (69 FR 68008), which became effective on December 22, 2004. The provisions of 10 CFR 50.69 allow for the adjustment of the scope of SSCs subject to special treatment requirements. Special treatment refers to those requirements that provide increased assurance beyond normal industry practices that SSCs perform their design basis functions. For SSCs categorized as low safety significance (LSS), alternative treatment requirements may be implemented in accordance with the regulation. For SSCs determined to be of high safety significance (HSS), requirements may not be changed.

Section 50.69 of 10 CFR contains requirements regarding how a licensee categorizes SSCs using a risk-informed process, adjusts treatment requirements consistent with the relative significance of the SSC, and manages the process over the lifetime of the plant. A risk-informed categorization process is employed to determine the safety significance of SSCs and place the SSCs into one of four risk-informed safety class (RISC) categories. The determination of safety significance is performed by an integrated decision-making process, which uses both risk insights and traditional engineering insights. The safety functions include the design-basis functions, as well as functions credited for severe accidents (including external events). Special or alternative treatment for the SSCs is applied as necessary to maintain functionality and reliability and is a function of the 10 CFR 50.69 categorization process results and associated bases. Finally, periodic assessment activities are conducted to adjust the categorization and/or treatment processes as needed so that SSCs continue to meet all applicable functional requirements.

Section 50.69 of 10 CFR does not allow for the elimination of SSC functional requirements or allow equipment that is required by the deterministic design basis to be removed from the facility. Instead, 10 CFR 50.69 enables licensees to focus their resources on SSCs that make a significant contribution to plant safety. In 2004, when promulgating the 10 CFR 50.69 rule[2], the Commission stated:

> It is important to note that this rulemaking effort, while intended to ensure that the scope of special treatment requirements imposed on SSCs is risk-informed, is not intended to allow for the elimination of SSC functional requirements or to allow equipment that is required by the deterministic design basis to be removed from the facility (i.e., changes to the design of the facility must continue to meet

[2] Final Rule, Risk-Informed Categorization and Treatment of Structures, Systems and Components for Nuclear Power Reactors, 69 Fed. Reg. 68008, 68011 (Nov. 22, 2004).

> the current requirements governing design change; most notably § 50.59). Instead, this rulemaking should enable licensees and the staff to focus their resources on SSCs that make a significant contribution to plant safety by restructuring the regulations to allow an alternative risk-informed approach to special treatment. Conversely, for SSCs that do not significantly contribute to plant safety on an individual basis, this approach should allow an acceptable, though reduced, level of confidence (i.e., "reasonable confidence") that these SSCs will satisfy functional requirements. However, continued maintenance of the health and safety of the public will depend on effective implementation of § 50.69 by the licensee or applicant applying the rule at its nuclear power plant.

For SSCs that are categorized as HSS, existing treatment requirements are maintained or potentially enhanced. Conversely, for SSCs categorized as LSS that do not significantly contribute to plant safety on an individual basis, the regulation allows an alternative risk-informed approach to treatment that provides a reasonable level of confidence that these SSCs will satisfy functional requirements. Implementation of 10 CFR 50.69 allows licensees to improve focus on equipment that has HSS.

2.2 Licensee Proposed Changes

In its letter dated March 21, 2019 (Reference 2), the licensee proposed to amend the Sequoyah RFOLs by adding the following license condition that would allow for the implementation of 10 CFR 50.69:

> Adoption of 10 CFR 50.69, "Risk-Informed categorization and treatment of structures, systems and components for nuclear power plants"
>
> (1) TVA is approved to implement 10 CFR 50.69 using the processes for categorization of Risk Informed Safety Class (RISC)-1, RISC-2, RISC-3, and RISC-4 structures, systems, and components (SSCs) using: Probabilistic Risk Assessment (PRA) model to evaluate risk associated with internal events, including internal flooding; using the fire safe shutdown equipment list in the SQN Fire Protection Report referenced in the Updated Final Safety Analysis Report to evaluate internal fire events; the NUMARC 91-06 shutdown safety assessment process to assess shutdown risk; the Arkansas Nuclear One, Unit 2 (ANO-2) passive categorization method to assess passive component risk for Class 2 and Class 3 SSCs and their associated supports; and the results of non PRA evaluations that are based on the IPEEE Screening Assessment for External Hazards, i.e., seismic margin analysis (SMA) to evaluate seismic risk, and a screening of other external hazards updated using the criteria in the endorsed ASME/ANS RA-Sa-2009 PRA Standard for other external hazard screening significance; as specified in Unit 1 [Unit 2] License Amendment [Number].
>
> (2) Prior to implementation of the provisions of 10CFR 50.69, TVA shall complete the items below;
>
> a. Items listed in Enclosure 1, Attachment 1, "SQN 10 CFR 50.69 PRA Implementation Items," in TVA letter CNL-19-002, "Response to Request for Additional Information Regarding Application to Modify Sequoyah Nuclear Plant Units 1 and 2, Application to Adopt 10 CFR 50.69,

"Risk-informed Categorization and Treatment of Structures, Systems, and Components for Nuclear Power Reactors, (SQN-TS-17-06) (EPID: L-2018-LLA-0066)," dated March 21, 2019.

(3) Prior NRC approval, under 10 CFR 50.90, is required for a change to the categorization process specified above (e.g., change from a seismic margins approach to a seismic probabilistic risk assessment approach, change from alternative method for internal fire to a fire probabilistic risk assessment approach).

2.3 Regulatory Review

The NRC staff considered the following regulatory requirements and guidance during its review of the proposed changes.

Regulatory Requirements

Section 50.69 of 10 CFR provides an alternative approach for establishing requirements for treatment of SSCs for nuclear power reactors using an integrated and systematic risk-informed process for categorizing SSCs according to their safety significance. Specifically, for SSCs categorized as LSS, alternative treatment requirements may be implemented in accordance with the regulation. For SSCs determined to be of HSS, requirements may not be changed.

Paragraph 50.69(c) of 10 CFR requires the licensees to use an integrated decision-making process to categorize safety-related and non-safety-related SSCs according to the safety significance of the functions they perform into one of the following four RISC categories, which are defined in 10 CFR 50.69(a), as follows:

RISC-1: Safety-related SSCs that perform safety significant functions[3]

RISC-2: Nonsafety-related SSCs that perform safety significant functions

RISC-3: Safety-related SSCs that perform LSS functions

RISC-4: Nonsafety-related SSCs that perform LSS functions

The SSCs are classified as having either HSS functions (i.e., RISC-1 and RISC-2 categories) or LSS functions (i.e., RISC-3 and RISC-4 categories). For HSS SSCs, 10 CFR 50.69 maintains current regulatory requirements (i.e., it does not remove any requirements from these SSCs) for special treatment. For LSS SSCs, licensees can implement alternative treatment requirements in accordance with 10 CFR 50.69(b)(1) and 10 CFR 50.69(d). For RISC-3 SSCs, licensees can replace special treatment with an alternative treatment. For RISC-4 SSCs, 10 CFR 50.69 does not impose new treatment requirements, and RISC-4 SSCs are removed from the scope of any applicable special treatment requirements identified in 10 CFR 50.69(b)(1).

[3] Nuclear Energy Institute (NEI) 00-04, Revision 0, "10 CFR 50.69 SSC Categorization Guideline," July 2005 (Reference 5), uses the term "high-safety-significant" to refer to SSCs that perform safety-significant functions. The NRC understands HSS to have the same meaning as "safety-significant" (i.e., SSCs that are categorized as RISC-1 or RISC-2), as used in 10 CFR 50.69.

Paragraph 50.69(c)(1) of 10 CFR states that SSCs must be categorized as RISC-1, RISC-2, RISC-3, or RISC-4 SSCs using a categorization process that determines if an SSC performs one or more safety-significant functions and identifies those functions. The process must:

(i) Consider results and insights from the plant-specific PRA. This PRA must, at a minimum, model severe accident scenarios resulting from internal initiating events occurring at full power operation. The PRA must be of sufficient quality and level of detail to support the categorization process and must be subjected to a peer review process assessed against a standard or set of acceptance criteria that is endorsed by the NRC.

(ii) Determine SSC functional importance using an integrated, systematic process for addressing initiating events (internal and external), SSCs, and plant operating modes, including those not modeled in the plant-specific PRA. The functions to be identified and considered include design bases functions and functions credited for mitigation and prevention of severe accidents. All aspects of the integrated, systematic process used to characterize SSC importance must reasonably reflect the current plant configuration and operating practices, and applicable plant and industry operational experience.

(iii) Maintain defense-in-depth [(DID)].

(iv) Include evaluations that provide reasonable confidence that for SSCs categorized as RISC-3, sufficient safety margins are maintained and that any potential increases in core damage frequency (CDF) and large early release frequency (LERF) resulting from changes in treatment permitted by implementation of §§ 50.69(b)(1) and (d)(2) are small.

(v) Be performed for entire systems and structures, not for selected components within a system or structure.

Paragraph 50.69(c)(2) of 10 CFR states:

> The SSCs must be categorized by an Integrated Decision-Making Panel (IDP) staffed with expert, plant-knowledgeable members whose expertise includes, at a minimum, PRA, safety analysis, plant operation, design engineering, and system engineering.

Paragraph 50.69(b)(3) of 10 CFR states that the Commission will approve a licensee's implementation of this section by issuance of a license amendment if the Commission determines that the categorization process satisfies the requirements of 10 CFR 50.69(c). As stated in 10 CFR 50.69(b), after the NRC approves an application for a license amendment, a licensee may voluntarily comply with 10 CFR 50.69, as an alternative to compliance with the following requirements for LSS SSCs:

(i) 10 CFR Part 21,

(ii) A portion of 10 CFR 50.46a(b),

(iii) 10 CFR 50.49,

(iv) 10 CFR 50.55(e)

(v) Specified requirements of 10 CFR 50.55a

(vi) 10 CFR 50.65, except for paragraph (a)(4),

(vii) 10 CFR 50.72,

(viii) 10 CFR 50.73,

(ix) Appendix B, "Quality Assurance Criteria for Nuclear Power Plants and Fuel Reprocessing Plants," to 10 CFR Part 50,

(x) Specified requirements for containment leakage testing, and

(xi) Specified requirements of Appendix A, "Seismic and Geologic Siting Criteria for Nuclear Power Plants," to 10 CFR Part 100.

Guidance

Nuclear Energy Institute (NEI) 00-04, Revision 0, "10 CFR 50.69 SSC Categorization Guideline," July 2005 (Reference 5), describes a process for determining the safety significance of SSCs and categorizing them into the four RISC categories defined in 10 CFR 50.69. This categorization process is an integrated decision-making process that incorporates risk and traditional engineering insights. Revision 0 of NEI 00-04 provides options for licensees implementing different approaches depending on the scope of their PRA models. It also allows for the use of non-PRA approaches when PRAs models have not been developed. The NEI 00-04 guidance identifies non-PRA methods to be used as an approach, such as fire-induced vulnerability evaluation (FIVE) to address internal fire risk, seismic margin analysis (SMA) to address seismic risk, and guidance in the Nuclear Management and Resources Council (NUMARC) report NUMARC 91-06, "Guidelines for Industry Actions to Assess Shutdown Management," December 1991 (Reference 6), to address shutdown operations. As stated in Regulatory Guide (RG) 1.201, Revision 1, "Guidelines for Categorizing Structures, Systems, and Components in Nuclear Power Plants According to their Safety Significance," May 2006 (Reference 4), such non-PRA-type evaluations will result in more conservative categorization, in that special treatment requirements will not be allowed to be relaxed for SSCs that are relied upon in such evaluations. The degree of relief that the NRC will accept under 10 CFR 50.69 (i.e., SSCs subject to relaxation of special treatment requirements) will be commensurate with the assurance provided by the evaluations performed to assess and characterize the SSC's risk.

Sections 2 through 10 of NEI 00-04, Revision 0, guidance describes steps/elements of the SSC categorization process to be performed for meeting the requirements of 10 CFR 50.69, as follows:

- Sections 3.2 and 5.1 provide specific guidance corresponding to 10 CFR 50.69(c)(1)(i).
- Sections 3, 4, 5, and 7 provide specific guidance corresponding to 10 CFR 50.69(c)(1)(ii).

- Section 6 provides specific guidance corresponding to 10 CFR 50.69(c)(1)(iii).
- Section 8 provides specific guidance corresponding to 10 CFR 50.69(c)(1)(iv).
- Section 2 provides specific guidance corresponding to 10 CFR 50.69(c)(1)(v).
- Sections 9 and 10 provide specific guidance corresponding to 10 CFR 50.69(c)(2).

Additionally, Section 11 of NEI 00-04, Revision 0, provides guidance on program documentation and change control related to the requirements of 10 CFR 50.69(e). Section 12 of NEI 00-04, Revision 0, provides guidance on periodic review related to the requirements in 10 CFR 50.69(f). Maintaining change control and periodic review provides confidence that all aspects of the program reasonably reflect the current as-built and as-operated plant configuration and operating practices, and applicable plant and industry operational experience, as required by 10 CFR 50.69(c)(1)(ii).

RG 1.201, Revision 1 (Reference 4), endorses the categorization process described in NEI 00-04, Revision 0 (Reference 5), with clarifications, limitations, and conditions. RG 1.201, Revision 1, states that the applicant is expected to document, at a minimum, the technical adequacy of the internal initiating events PRA. Licensees may use either PRAs or alternative approaches for hazards other than internal initiating events. RG 1.201, Revision 1, clarifies that the NRC staff expects that licensees proposing to use non-PRA approaches in their categorization should provide a basis in the submittal for why the approach, and the accompanying method employed to assign safety significance to SSCs, is technically acceptable. The guidance further states that as part of the NRC's review and approval of a licensee's or applicant's application requesting to adopt 10 CFR 50.69, the NRC staff intends to impose a license condition that will explicitly address the scope of the PRA and non-PRA methods used in the licensee's categorization approach. If a licensee or applicant wishes to change its categorization approach and the change is outside the bounds of the NRC's license condition (e.g., switch from an SMA to a seismic PRA), the licensee or applicant will need to seek NRC approval via a license amendment of the implementation of the new approach or method for use in its categorization process. In addition, RG 1.201, Revision 1, states that all aspects of NEI 00-04, Revision 0, should be followed to achieve reasonable confidence in the evaluations required by 10 CFR 50.69(c)(1)(iv).

RG 1.200, Revision 2, "An Approach for Determining the Technical Adequacy of Probabilistic Risk Assessment Results for Risk-Informed Activities," March 2009 (Reference 7), describes an acceptable approach for determining whether the acceptability of the base PRA, in total or the parts, that is used to support an application, is sufficient to provide confidence in the results, such that the PRA can be used in regulatory decisionmaking for light-water reactors. It endorses, with clarifications, the American Society of Mechanical Engineers (ASME)/American Nuclear Society (ANS) PRA standard, ASME/ANS RA-Sa-2009, "Standard for Level 1/Large Early Release Frequency Probabilistic Risk Assessment for Nuclear Power Plant Applications," February 2009 (Reference 8). Revision 2 of RG 1.200 provides guidance for determining the acceptability of a PRA by comparing the PRA to the relevant parts of ASME/ANS RA-Sa-2009 using a peer review process. The guidance discusses the need to perform peer reviews for PRA upgrades. A PRA upgrade is defined in ASME/ANS RA-Sa-2009 as "the incorporation into a PRA model of a new methodology, or significant changes in scope or capability that impact the significant accident sequences or the significant accident progression sequences."

RG 1.174, Revision 3, "An Approach for Using Probabilistic Risk Assessment in Risk-Informed Decisions on Plant-Specific Changes to the Licensing Basis," January 2018 (Reference 9), provides guidance on the use of PRA findings and risk insights in support of changes to a plant's licensing basis. Revision 3 of RG 1.174 provides risk acceptance guidelines for evaluating the results of such evaluations.

NUREG-1855, Revision 1, "Guidance on the Treatment of Uncertainties Associated with PRAs in Risk-Informed Decisionmaking," March 2017 (Reference 10), provides guidance on how to treat uncertainties associated with PRA in risk-informed decisionmaking. The guidance fosters an understanding of the uncertainties associated with PRA and their impact on the results of the PRA and provides a pragmatic approach to addressing these uncertainties in the context of the decisionmaking.

3.0 TECHNICAL EVALUATION

3.1 Method of NRC Staff Review

The NRC staff reviewed the licensee's 10 CFR 50.69 categorization process against the categorization process described in NEI 00-04, Revision 0 (Reference 5), as endorsed in RG 1.201, Revision 1 (Reference 4), and the acceptability of the licensee's PRA for use in the application of the 10 CFR 50.69 categorization process. The NRC staff's review, as documented in this safety evaluation (SE), uses the framework provided in NEI 00-04, Revision 0, as endorsed in RG 1.201, Revision 1.

3.2 Overview of the Categorization Process (NEI 00-04, Revision 0, Section 2)

Section 2 of NEI 00-04, Revision 0, in part, states that the categorization process includes eight primary steps:

1. Assembly of Plant-Specific Inputs (Section 3 of NEI 00-04, Revision 0)
2. System Engineering Assessment (Section 4 of NEI 00-04, Revision 0)
3. Component Safety Significance Assessment (Section 5 of NEI 00-04, Revision 0)
4. Defense-In-Depth Assessment (Section 6 of NEI 00-04, Revision 0)
5. Preliminary Engineering Categorization of Functions (Section 7 of NEI 00-04, Revision 0)
6. Risk Sensitivity Study (Section 8 of NEI 00-04, Revision 0)
7. IDP Review and Approval (Section 9 of NEI 00-04, Revision 0)
8. SSC Categorization (Section 10 of NEI 00-04, Revision 0)

The licensee stated in Section 3.1.1 of the LAR (Reference 1) that it will implement the risk categorization process in accordance with NEI 00-04, Revision 0 (Reference 5), as endorsed in RG 1.201, Revision 1 (Reference 4).

The licensee provided further discussion of specific elements within the 10 CFR 50.69 categorization process. Elements of the categorization process for which the licensee provided clarity in the LAR are bulleted below. A more detailed review of those specific elements in the categorization process is discussed in the applicable sections of this SE.

- *Passive Characterization*: Passive components are not modeled in the PRA. Therefore, a different method to perform the assessment is used to assess the safety significance of these components, as described in Section 3.5.3.5 of this SE. The process used addresses those components that have only a pressure retaining function and the passive function of active components, such as the pressure/liquid retention of the body of a motor-operated valve.

- *Qualitative Characterization:* System functions are qualitatively categorized as HSS or LSS based on the seven questions in Section 9.2 of NEI 00-04, Revision 0 (Refer to Section 3.5.8 of this SE).

- *Cumulative Risk Sensitivity Study:* For PRA-modeled components, an overall risk sensitivity study is used to confirm that the population of LSS components results in acceptably small increases to CDF and LERF and meets the acceptance guidelines of RG 1.174, Revision 3 (Reference 9) (Refer to Section 3.5.7 of this SE).

- *Review by the IDP:* The categorization results are presented to the IDP for review and approval. The IDP reviews the categorization results and makes the final determination on the safety significance of system functions and components (Refer to Section 3.5.8 of this SE).

- *Use of Fire Safe Shutdown Equipment List (FSSEL):* Sequoyah has proposed the use of the FSSEL to assess the fire risk. The use of this method is a deviation from the approaches and methodologies described in NEI 00-04, Revision 0 (Reference 5) (Refer to Section 3.5.3.1 of this SE).

Attachment 1 of Enclosure 1 to the licensee's letter dated March 21, 2019 (Reference 2), provides a list of steps/elements that the licensee will include in the 10 CFR 50.69 programmatic procedures prior to the use of the categorization process. A more detailed review of the steps/elements for the programmatic procedures is discussed in Section 3.10 of this SE.

3.3 Assembly of Plant-Specific Inputs (NEI 00-04, Revision 0, Section 3)

Paragraph 50.69(c)(1)(ii) of 10 CFR requires licensees to determine SSC functional importance using an integrated, systematic process for addressing initiating events (i.e., internal and external), SSCs, and plant operating modes, including those not modeled in the plant-specific PRA. The functions to be identified and considered include design-basis functions and functions credited for mitigation and prevention of severe accidents.

Section 3 of NEI 00-04, Revision 0 (Reference 5), states that the assembly of plant-specific inputs involves the collection and assessment of the key inputs to the risk-informed categorization process. This includes design and licensing information, PRA analyses, and other relevant plant data sources. This step also includes the critical evaluation of plant-specific risk information to ensure that the PRA is modeled adequately to support the risk characterization of SSCs for this application. In addition, Section 4 of the NEI 00-04 guidance states, in part, "that the next step is the identification of system functions, including design-basis and beyond-design-basis functions identified in the PRA, and that system functions should be consistent with the functions defined in design-basis documentation and maintenance rule functions."

Furthermore, the guidance in Section 3 of NEI 00-04, Revision 0, summarizes the use of PRA, if such PRA models exist, or, in the absence of quantifiable PRA, the use of other methods to evaluate risk including the FIVE methodology, SMA, Individual Plant Examination of External Events (IPEEE) Screening, and Shutdown Safety Plan. The NRC staff acknowledges that elements of the categorization process are not always performed in chronological order and may be performed in parallel, such that, the systematic process for evaluating the plant-specific PRA may include other aspects of the categorization process (e.g., system selection, system boundary definition, identification of system functions, and mapping of components to functions) that are further discussed in Section 3.4 of this SE. The licensee's risk categorization process uses PRAs to assess risks from the internal events PRA (IEPRA) (includes internal floods). For the other applicable risk hazard groups, the licensee's process uses non-PRA methods for risk characterization, including the Sequoyah SMA (Reference 11) to assess seismic risk from the IPEEE results, Sequoyah FSSEL to assess fire risk, and guidance provided in NEI 00-04 to assess the risk from external hazards (e.g., high winds, external floods) and other hazards.[4] To assess risk from shutdown operations, the licensee's categorization process uses the Shutdown Safety Management Plan. The NRC staff review of the quality and level of detail for the acceptability of the IEPRA at the time of the submittal and non-PRA methods is provided in Sections 3.5.1 through 3.5.3 of this SE.

Paragraph 50.69(c)(1)(v) of 10 CFR requires that SSC categorization be performed for entire systems and structures, not for selected components within a system or structure. The NRC staff finds the process described in the LAR, as supplemented by letter dated March 21, 2019 (Reference 2), is consistent with NEI 00-04, as endorsed by the staff in RG 1.201, Revision 1, and capable of collecting and organizing information at the system level for defining boundaries, functions, and components.

3.4 System Engineering Assessment (NEI 00-04, Revision 0, Section 4)

Paragraph 50.69(c)(1)(ii) of 10 CFR requires, in part, the functions to be identified and considered in the categorization process include design basis functions and functions credited for mitigation and prevention of severe accidents. Revision 0 of NEI 00-04 (Reference 5), includes guidance to identify all functions performed by each system and states that the IDP will categorize all system functions. All system functions include all functions involved in the prevention and mitigation of accidents and may include additional functions not credited as hazard mitigating functions depending on the system. The guidance in NEI 00-04 also includes consideration of interfacing functions. Section 4 of NEI 00-04 provides guidance for circumstances when the categorization of a candidate low safety-significant SSC within the scope of the system being considered cannot be completed because it also supports an interfacing system. The guidance states, in part, "[i]n this case, the SSC will remain uncategorized until the interfacing system is considered ... Therefore, the SSC will remain uncategorized and continue to receive its current level of treatment requirements." Furthermore, Section 7.1 of the NEI 00-04 guidance states, in part, "[d]ue to the overlap of

[4] Other hazards include any internal or external hazards that are not considered as part of the development of an internal events, internal flood, internal fire, seismic, high wind, or external flood PRA using the applicable parts of the ASME/ANS PRA standard, as endorsed by the NRC.

functions and components, a significant number of components support multiple functions. In this case, the SSC or part thereof should be assigned the highest risk significance for any function that the SSC or part thereof supports."

In Section 2.2 of the LAR (Reference 1), the licensee stated "[t]he safety functions in the categorization process include the design basis functions, as well as functions credited for severe accidents (including external events)." Section 3.1.1 of the LAR summarizes the different hazards and plant states for which functional and risk significant information will be collected. In Section 3.1.1 of the LAR, the licensee confirmed that the SSC categorization process documentation will include, among other items, system functions, identified and categorized with the associated bases, and mapping of components to support function(s).

Therefore, the NRC staff finds that the process described in the LAR is consistent with NEI 00-04, Revision 0, as endorsed by the NRC in RG 1.201, Revision 1, and meets the requirements set forth in paragraph 50.69(c)(1)(ii).

3.5 Component Safety Significance Assessment (NEI 00-04, Section 5)

This step in the licensee's categorization process assesses the safety significance of components using quantitative or qualitative risk information from a modeled PRA hazard, other hazards that can be screened, and non-PRA method(s). In the NEI 00-04 guidance (Reference 5), component risk significance is assessed separately for the following hazard groups:

- Internal Events (includes internal floods)
- Internal Fire Events[5]
- Seismic Events
- External Hazards (e.g., high winds, external floods)
- Other Hazards
- Shutdown Events
- Passive Categorization[6]

Paragraph 50.69(c)(1)(i) of 10 CFR requires, in part, the use of PRA to assess risk from internal events as a minimum. This paragraph of the rule further specifies that the PRA used in the categorization process must be of sufficient quality and level of detail and subject to an acceptable peer review process. Paragraph 50.69(b)(2) of 10 CFR allows, and the guidance in NEI 00-04, Revision 0 (Reference 5) summarizes, the use of PRA, if such PRA models exist, or, in the absence of quantifiable PRA, the use of other methods (e.g., FIVE, SMA, IPEEE, screening, and shutdown safety management plan).

In Sections 3.1.1 and 3.2.1 of the LAR (Reference 1), the licensee described that the Sequoyah categorization process uses PRA modeled hazards to assess risks for the internal events (includes internal flood). For the other risk contributors, the licensee's process uses the following non-PRA methods to characterize the risk:

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[5] Deviation: Methodology proposed for use of the FSSEL to assess the risk for internal fire events not endorsed by the NRC in RG 1.201, Revision 1 (Reference 4).

[6] Deviation: Methodology proposed for the categorization of passive components not cited in NEI 00-04, Revision 0 (Reference 5), or RG 1.201, Revision 1 (Reference 4), but approved by the NRC for Arkansas Nuclear One, Unit 2 (ANO-2) for issuance of another license amendment (Reference 12).

- Seismic: SMA performed for the Sequoyah IPEEE (Reference 11).

- Internal Fire Events: Living fire safe shutdown equipment list (FSSEL) performed for the Sequoyah Fire Protection program.

- External Hazards: Screening analysis performed for IPEEE (Reference 11) updated using criteria from Part 6 of the ASME/ANS RA-Sa-2009 PRA Standard, as endorsed by the NRC.

- Other Hazards: Screening analysis performed for the IPEEE (Reference 11) updated using criteria from Part 6 of the ASME/ANS RA-Sa-2009 PRA Standard, as endorsed by the NRC.

- Shutdown Events: Safe Shutdown Risk Management program consistent with NUMARC 91-06 (Reference 6).

- Passive Components: ANO-2 passive categorization methodology (Reference 12)

The approaches and methods proposed by the licensee to address internal events, seismic, external events, other hazards, DID, and shutdown events are consistent with the approaches and methods included in the guidance in NEI 00-04, Revision 0 (Reference 5), as endorsed in RG 1.201, Revision 1 (Reference 4). The non-PRA method for the categorization of passive components is consistent with the ANO-2 methodology for passive components (Reference 12) approved for risk-informed safety classification and treatment for repair/replacement activities in Class 2 and 3 moderate and high energy systems. A detailed staff review for the use of the ANO-2 method in the SSC categorization process is provided in Section 3.5.3.5 of this SE. To address internal fire events, the licensee proposed to use an alternative method not specified in the NEI 00-04 guidance as endorsed by the NRC (Reference 5). A detailed staff review of the licensee's proposed approach for the use of the FSSEL is provided in Section 3.5.3.1 of this SE.

3.5.1 Evaluation of PRA Acceptability to Support the SSC Categorization Process

Consistent with Section 5 of NEI 00-04, Revision 0 (Reference 5), the component safety significance assessment must include evaluations for each of the hazards : (1) internal events hazard, (2) internal fire hazard, (3) seismic hazard, (4) other hazards (e.g., high wind, external floods, etc.), and (5) shutdown events.

Paragraph 50.69(c)(i) of 10 CFR requires, in part, a licensee's PRA must be of sufficient quality and level of detail to support the categorization process and must be subjected to a peer review process assessed against a standard or set of acceptance criteria that is endorsed by the NRC. 10 CFR 50.69(b)(2)(iii) further requires that the results of the peer review process conducted to meet 10 CFR 50.69(c)(1)(i) criteria be submitted as part of the application.

3.5.1.1 Scope of PRA

The Sequoyah PRA is comprised of a full-power, Level 1, IEPRA, which evaluates the CDF and LERF risk metrics.

The licensee discussed in Section 3.3 of the LAR (Reference 1) that the IEPRA (includes internal floods) model has been assessed against RG 1.200, Revision 2 (Reference 7).

Furthermore, LAR Section 3.3 states that a finding closure review was conducted on the identified PRA model on May 8 to May 10, 2017, using the process. Closed findings were reviewed and closed using the NRC-accepted process documented in the NEI letter to the NRC "Final Revision of Appendix X to NEI 05-04/07-12/12-16, 'Close-out of Facts and Observations,'" dated February 21, 2017 (Reference 13). The NRC staff finds that the information provided in the LAR to support the staff's review of the IEPRA (includes internal floods) for technical acceptability was sufficient and therefore meets the requirements set forth in 10 CFR 50.69(b)(2)(iii).

Aspects considered by the staff to evaluate the scope of the PRA include: (1) peer review history, (2) the Appendix X, Independent Assessment process, (3) credit for FLEX in the PRA, and (4) assessment of assumptions and approximations. In e-mail correspondence to the licensee on January 15, 2019 (Reference 3), the NRC staff issued RAIs to further assess the acceptability of Sequoyah's IEPRA (includes internal floods) for consistency with RG 1.200, Revision 2 (Reference 7), and NEI 00-04, Revision 0 (Reference 5), as endorsed in RG 1.201, Revision 1 (Reference 4). The staff's review of these aspects of the PRA and supplemental responses to the applicable RAIs are provided in subsections 3.5.1.2 through 3.5.1.5 of this SE.

3.5.1.2 Internal Events PRA (Includes Internal Floods) Peer Review History

In Section 3.3 of the LAR, the licensee stated that "the internal events PRA model was subjected to a self-assessment and a full-scope peer review in January 2011." Subsequently, in May 2017, TVA performed an Independent Assessment for closure of the finding-level F&Os and concluded all the IEPRA (includes internal floods) F&Os has been closed. A detailed staff review of this May 2017 Independent Assessment is included below, in Section 3.5.1.3 of this SE.

In Section 3.2 of the LAR, for the IEPRA (includes internal floods), TVA stated, in part, "there are no PRA upgrades that have not been peer reviewed." Section X.1.3 of Appendix X to NEI 05-04, 07-12, and 12-13,[7] as accepted by NRC staff in a memorandum dated May 3, 2017,[8] provides guidance that includes a written assessment and justification of whether the resolution of each F&O, within the scope of the Independent Assessment, constitutes a PRA upgrade or maintenance update as defined in the ASME/ANS Ra-Sa-2009 PRA Standard (Reference 8). The staff performed an electronic audit and followed up with RAI 01.a through d (Reference 3) requesting the licensee confirm that the Independent Assessment performed in May 2017 was performed consistent with the NEI Appendix X guidance, as accepted, with conditions, by the NRC staff. In its review of the licensee's response to RAI 01.a (Reference 3) in Section 3.5.1.3 of this SE, the NRC staff concluded that all F&Os were appropriately assessed by the Independent Assessment team to assure that no new methods and/or upgrades were inadvertently incorporated into the IEPRA without a peer review in accordance with the ASME/ANS RA-Sa-2009 PRA standard as endorsed by the NRC. Therefore, the NRC staff finds that the Sequoyah IEPRA (includes internal floods) was appropriately peer reviewed consistent with RG 1.200, Revision 2 and meets the requirements set forth in 10 CFR 50.69(c)(1)(i).

[7] *Errata;* Anderson, V. K., Nuclear Energy Institute, letter to Stacey Rosenberg, U.S. Nuclear Regulatory Commission, "Final Revision of Appendix X to NEI 05-04/07-12/12-16, 'Close-Out of Facts and Observations,'" dated February 21, 2017 (Reference 13).

[8] Giitter, J., and Ross-Lee, M. J., U.S. Nuclear Regulatory Commission, letter to Mr. Greg Krueger, Nuclear Energy Institute, "U.S. Nuclear Regulatory Commission Acceptance on Nuclear Energy Institute Appendix X to Guidance 05-04, 07-12, and 12-13, Close-Out of Facts and Observations (F&Os)," dated May 3, 2017 (Reference 14).

3.5.1.3 Appendix X, Independent Assessment Process for F&O Closure

Section X.1.3 of Appendix X to NEI 05-04, NEI 07-12, and NEI 12-13 (Reference 13) provide guidance to perform an Independent Assessment for the closure of F&O(s) identified from a full-scope or focused-scope peer review. Appendix X includes guidance for the Independent Assessment process regarding: (i) the qualifications of the Independent Assessment team members, (ii) pre-review activities, (iii) on-site review activities, and (iv) post-review activities, thus assuring that closure of the F&Os are met at capability category (CC) II for the applicable supporting requirements (SR) at in the ASME/ANS Ra-SA-2009 PRA Standard (Reference 8), as endorsed by RG 1.200 Revision 2 (Reference 7).

LAR Section 3.3 (Reference 1) states, in part, "[a] finding closure review was conducted on the identified PRA models on May 8 to May 10, 2017. Closed findings were reviewed and closed using the process documented in the NEI letter to the NRC "Final Revision of Appendix X to NEI 05-04/07-12/12-16, Close-Out of Facts and Observations (F&O)," [...], as accepted by NRC..." In RAI 01 (Reference 3), the NRC staff requested the licensee provide additional information to confirm that the Independent Assessment for the IEPRA (which includes internal floods) was performed consistent with the process, as accepted, with conditions, by the NRC staff in letter dated May 3, 2017 (Reference 14). As a result and in response to RAI 01.a (Reference 2), TVA stated, in part, "[t]he Independent Assessment Team was not provided with a written assessment of whether the F&O resolutions constituted a PRA upgrade or a maintenance update as part of the F&O Closure process defined by Appendix X to NEI 05-04/07-12/12-16. In response to RAI 01.b and 01.d (Reference 2), TVA confirmed that a subsequent review of the F&O resolutions was performed in December 2018. In further response to RAI 01.d, TVA stated, "a specific evaluation was also provided for each closed F&O to document whether the review team considered the F&O resolution a "PRA Maintenance Update" or a "PRA upgrade." Upon completion of the subsequent review performed by the Independent Assessment team, no F&Os remained open." Thus, the NRC staff finds that upon the completion of the subsequent review performed by the Independent Assessment team, assessment for whether the closure of the F&O constituted an upgrade or a maintenance update has been completed consistent with the staff acceptance, with conditions, of Appendix X to NEI 05-04, NEI 07-12, and NEI 12-13 (Reference 13).

In RAI 01.d (Reference 3), the NRC further requested the licensee confirm how the Independent Assessment team assured that the aspects of the underlying SR that were previously not met, or met at Capability Category (CC)-I, are now met at CC-II. In response to the RAI (Reference 2) the licensee confirmed that the subsequent review also included documenting for each closed F&O that the resolution met CC-II requirements for the SRs that were identified as applicable to the F&O. Thus, the NRC staff finds that the licensee added additional documentation during a subsequent Independent Assessment review to clarify that the F&Os resolutions were reviewed and determined to now meet CC-II for the applicable SR(s).

Appendix X guidance states, in part:

> The relevant PRA documentation should be complete and have been incorporated into the PRA model and supporting documentation prior to closing the finding." [For closure of F&O(s) after the on-site review, Appendix X guidance explicitly states,] "[t]he host utility may, in the time between the on-site review and the finalization of the Independent Assessment team report, demonstrate that the issue has been addressed, that a closed finding has been

achieved, and that the documentation has been formally incorporated in the PRA Model of Record [MOR].

In RAI 01.b (Reference 3), the NRC staff requested the licensee confirm that all model changes associated with the closure of all F&Os reviewed during the Independent Assessment performed in May 2017 were incorporated into the IEPRA (includes internal floods) and/or the supporting documentation at the time of the finalization of the Independent Assessment team report. In response to RAI 01.b (Reference 2), the licensee stated that the F&O resolutions were not incorporated into the MOR; however, the F&O closure report documented that the changes initiated to resolve the F&Os reviewed by the Independent Assessment team have been incorporated into the living model and associated documentation.

Regulatory Guide 1.200, Revision 2 (Reference 7) does not provide specific guidance on the configuration control of the base PRA for a specific application. However, it does acknowledge that application-specific PRA models exists. Section 3.2 of the NEI 00-04 guidance (Reference 5) describes, in part, "an essential element of the SSC categorization process is a plant-specific full power internal events PRA, which should...reflect the as-built and as-operated plant..." The living PRA model includes the maintenance updates incorporated into the plant to represent the as-built, as-operated plant. As a part of the license condition for implementation of the 10 CFR 50.69 SSC categorization process provided in Enclosure 2 of the letter dated March 21, 2019 (Reference 2), TVA committed to update the IEPRA (includes internal floods) MOR with all the changes reviewed by the Independent Assessment team and documented in the Independent Assessment report for the resolution of F&Os performed in May 2017. The NRC staff finds that upon the completion of the license condition, the PRA MOR with the F&O resolutions will reflect the as-built-as-operated plant and the living PRA model reviewed by the Independent Assessment team and will therefore be consistent with NEI 00-04 guidance as endorsed.

In RAI 01.c (Reference 3), the NRC staff requested the licensee confirm if remote participation was used to conduct the Independent Assessment and include details for the NRC staff to confirm consistency with guidance provided in Appendix X. In response to RAI 01.c (Reference 2),.the licensee confirmed that the subsequent review was performed by the same individuals who performed the initial review of the F&Os, and this review was performed remotely with web-based consensus sessions. The NRC staff finds that the licensee used remote participation consistent with the guidance provided in Appendix X (Reference 13) as accepted by the NRC staff with conditions (Reference 14).

In review of the licensee's responses to RAI 01.a through d (Reference 2), the NRC staff concludes that the closure of F&Os for the IEPRA (includes internal floods) was performed consistently with Appendix X to NEI 05-04, 07-12, and 12-13 (Reference 13) as accepted, with conditions by the NRC staff and NEI 00-04 (Reference 5), as endorsed by the NRC for SSC categorization.

3.5.1.4 <u>Credit for FLEX Equipment</u>

The NRC memorandum dated May 30, 2017, "Assessment of the Nuclear Energy Institute 16-06, 'Crediting Mitigating Strategies in Risk-Informed Decision Making,' Guidance for Risk-Informed Changes to Plants Licensing Basis" (Reference 15), provides the NRC staff's assessment of challenges to incorporating FLEX equipment and strategies into a PRA model in support of risk-informed decision making in accordance with the guidance in RG 1.200, Revision 2 (Reference 7). In response to RAI 08 (Reference 2), the licensee stated that "[t]he

SQN PRA does not credit FLEX equipment or FLEX strategies in the PRA." The NRC staff finds that the TVA IEPRA (includes internal floods) does not credit FLEX equipment for the SSC categorization process; therefore, no new methods and/or upgrades were inadvertently incorporated into the Sequoyah IEPRA (includes internal floods) without a peer review consistent with the ASME/ANS RA-Sa-2009 PRA standard (Reference 8), as endorsed by the NRC.

3.5.1.5 Assessment of Assumptions and Approximations

Paragraphs 50.69(c)(1)(i) and (ii) of 10 CFR require that a licensee's PRA be of sufficient quality and level of detail to support the 10 CFR 50.69 categorization process, and that all aspects of the integrated, systematic process used to characterize SSC importance must reasonably reflect the current plant configuration and operating practices, and applicable plant and industry operational experience.

Table A-1 of RG 1.200, Revision 2, entitled, "Staff Position on ASME/ANS RA-Sa-2009 Part 1, General Requirements for an At-Power Level 1 and LERF PRA" (Reference 7), includes the staff clarification for Section 1-6.1 of ASME/ANS RA-Sa-2009 (Reference 8). The resolution for this clarification states, in part, "[t]herefore, the peer review shall also assess the appropriateness of the assumptions." Regulatory Guide 1.174, Revision 3 (Reference 9), cites NUREG-1855, Revision 1 (Reference 10), as related guidance that includes changes associated with expanding the discussion of sources of uncertainties. NUREG-1855, Revision 1, states, in part, "RG 1.200 [NRC 2009] and the PRA consensus standard published by ASME and ANS (ASME/ANS, 2009) each recognize the importance of identifying and understanding uncertainties as part of the process of achieving acceptability in a PRA, and these references provide guidance on this subject."

Section 3.2.7 of the LAR states that guidance in NUREG-1855, Revision 0, "Guidance on the Treatment of Uncertainties Associated with PRAs in Risk-Informed Decisionmaking," March 2009 (Reference 16), and Electric Power Research Institute (EPRI) TR-1016737, "Treatment of Parameter and Modeling Uncertainty for Probabilistic Risk Assessments," December 2008 (Reference 17), was used to identify, characterize, and screen model uncertainties. The NRC staff notes that EPRI TR-1026511, "Practical Guidance on the Use of Probabilistic Risk Assessment in Risk-Informed Application with a Focus on the Treatment of Uncertainty," December 2012 (Reference 18), is not included for consideration in NUREG-1855, Revision 0.

Identification and Characterization of Assumptions and Sources of Uncertainty

To assess the assumptions and sources of uncertainty identified by Sequoyah for the base IEPRA (includes internal floods), in RAI 04.a (Reference 3), the NRC staff requested the licensee provide discussion and justification of the process (e.g., Revision 0 or Revision 1) used to identify the key assumptions and sources of uncertainty provided in the LAR. In response to RAI 04.a (Reference 2), the licensee provided a summary on how Stage E of the guidance in NUREG-1855, Revision 1, was applied, which provides guidance on assessing model uncertainty. TVA stated, in part "[f]or SQN, this process was performed by reviewing PRA documentation for generic issues identified in Table A-1 of EPRI 1016737, as well as identifying plant-specific assumptions and uncertainties, and is therefore consistent with substep E-1.1 of NUREG-1855 Revision 1." Furthermore, Section 1.3 of NUREG-1855, Revision 1 (Reference 10) states, in part, "[a]lthough assumptions and approximations made on the level of detail in a PRA can influence the decisionmaking process, they are generally not considered to be model uncertainties because the level of detail in the PRA model could be enhanced, if necessary. Therefore, methods for identifying and characterizing issues associated with level of detail are not explicitly included in NUREG-1855; they are, however, addressed in EPRI reports 1016737 and 1026511." The NRC staff finds that the process used by the licensee to appropriately identify and characterize assumptions and sources of uncertainties for the base IEPRA (includes internal floods) using approved NRC guidance provided in EPRI TR-1016737 (Reference 17), is consistent with the guidance provided in NUREG-1855, Revision 1 (Reference 10).

3.5.1.6 Key Assumptions and Sources of Uncertainty for the Application

RG 1.200, Revision 2 (Reference 7), Section 3.3.2 provides guidance that states, in part, "[f]or each application that calls upon this regulatory guide, the applicant identifies the key assumptions and approximations relevant to that application. This will be used to identify sensitivity studies as input to the decision-making associated with the application."

Section 4.2 of RG 1.200, Revision 2, provides guidance that identifies specific information to be included in the licensee's submittal to demonstrate the technical adequacy of the PRA used in an application is of sufficient quality. The identified information includes the identification of the key assumptions and approximations relevant to the results used in the decision-making process.

Identification of Key Assumptions and Sources of Uncertainty

In RAI 04.a.(ii) (Reference 3), the NRC staff requested the licensee provide a brief description of how the key assumptions and sources of uncertainties provided in Attachment 6 of the LAR (Reference 1) were identified from the initial comprehensive list of PRA model(s) (i.e., base model) source of uncertainties. In response to RAI 04.a.(ii) (Reference 2), the licensee stated:

> This assessment is made by performing sensitivity analyses to determine the importance of the source of model uncertainty or related assumption to the acceptance criteria or guidelines. In the SQN uncertainty analysis, for any uncertainties that were not previously screened qualitatively, they were analyzed quantitatively in order to determine the impact on the PRA results. Uncertainties and assumptions that were identified as having an impact on the PRA results were then assessed to determine those applicable to the 10 CFR 50.69 application.

In response to RAI 04.a.(i) (Reference 2), TVA confirmed that substeps E-1.2 and E-1.3 of NUREG-1855, Revision 1 (Reference 10), were performed to identify, screen, and characterize those sources of model uncertainty and related assumptions in the base PRA that are relevant to the application. Substep E-1.4 of the guidance, TVA stated, is a qualitative screening process that involves identifying and validating whether consensus[9] models have been used in the PRA to evaluate identified model uncertainties. The licensee confirmed that for the SQN uncertainty analysis, some uncertainties and assumptions were screened based on the use of a consensus method.

[9] Per NUREG-1855, Revision 1, a consensus model is a model that has a publicly available published basis and has been peer reviewed and widely adopted by an appropriate stakeholder group.

In further response, the licensee confirmed that the results of the assessment were provided in Attachment 6 of the LAR, apart from the State of Knowledge Correlation (SOKC) uncertainty. The NRC staff finds that the assessment performed to identify the key assumptions/sources of uncertainty is consistent with the guidance provided in NUREG-1855, Revision 1.

<u>Treatment of the Key Assumptions and Sources of Uncertainty</u>

Section 5 of the NEI 00-04 guidance, as endorsed by the NRC, states, in part:

> An analysis of the impacts of parametric uncertainties on the importance measures used in this categorization process was performed and documented in EPRI TR-1008905, Parametric Uncertainty Impacts on Option 2 Safety Significance Categorization [...]. The conclusion of this analysis was that the importance measures used in combination with identified set of minimum sensitivity studies adequately address parametric uncertainties.

Furthermore, the guidance in NEI 00-04, Revision 0 (Reference 5), specifies sensitivity studies should be conducted for each PRA model to ensure that PRA assumptions and sources of uncertainty (e.g., human error, common cause failures, and maintenance probabilities) do not mask the SSC(s) importance. Table 5-2 of NEI 00-04 provides several recommended sensitivity studies for the internal events PRA. To assess how the dispositions provided in Attachment 6 of the LAR (Reference 1) concluded that the assumptions and sources of uncertainty provided in Attachment 6 of the LAR will be performed consistent with the NEI 00-04 guidance as endorsed by the NRC, in RAI 05 (Reference 3), the NRC staff requested the licensee provide which of the generic sensitivity studies outlined in Section 5 of the NEI 00-04 guidance is applicable for the following key assumptions/sources of uncertainty or if an additional "applicable sensitivity study" will be developed:

(1) In Attachment 6 of the LAR, the licensee identified a key assumption that a leaking pipe will be detected by visual inspection with a 0.9 probability. For the discussion of the key assumption, the licensee stated, in part, "[i]nternal flooding is a significant contributor to plant risk, and the detection probability is significant to reducing the impact of a flooding scenario." In response to RAI 05.a (Reference 2), the licensee summarized the development and use of probability of detection multipliers that reduce the frequency of pipe break initiating events based on different pipe sizes and types of detection and described how applicable sensitivity studies could be performed.

(2) In Attachment 6 of the LAR, the licensee did not provide a disposition for the key source of uncertainty regarding passive pipe break failures, human-induced flooding, and maintenance-induced flooding in the internal flooding PRA model. In response to RAI 05.b.(i) (Reference 2), for passive pipe breaks and human-induced flooding, the licensee clarified that the uncertainty associated with the passive pipe break failures are treated the same as the random uncertainty for other basic events. Therefore, the NRC staff concludes that the uncertainty in these frequencies would be included in the propagation of SOKC uncertainty throughout the PRA model. The licensee further clarified that the uncertainty associated with human-induced flooding events is treated the same as the other human reliability analysis actions within the model. Therefore, the NRC staff concludes that the uncertainty in these actions would be included in a generic sensitivity study increasing and decreasing all human error probabilities. To

address maintenance-induced flooding, in response to RAI 05.b.(ii), the licensee summarized how maintenance-induced flooding initiating events' frequencies are estimated and identified how applicable sensitivity studies could be performed.

(3) In Attachment 6 of the LAR (Reference 1), TVA identified a key assumption/ source of uncertainty that the type code data (i.e., used to represent modes of failure) for equipment include successful post-maintenance testing (PMT) and that this can result in an under-estimation of the failure probabilities. In response to RAI 05.c (Reference 2), the licensee summarized the collection, interpretation and use of PMT data, and identified how applicable sensitivity studies could be performed. In Attachment 6 of Enclosure 6 of the letter dated March 21, 2019 (Reference 2), for the disposition of equipment PMT data, the licensee discussed that, for the number of assumed PMT demands, the source of uncertainty had a small impact on CDF. However, for the inclusion of the number of successful PMT starts, the source of uncertainty could have a more profound effect on the CDF and LERF values.

NUREG-1855, Revision 1 (Reference 10), provides further guidance regarding how to address PRA uncertainties to assure the risk-informed decision making is in the context of the application for the decision under consideration. For the three key assumptions and sources of uncertainty described above, in response to RAI 05.a through c, the licensee confirmed that sensitivity studies will be performed consistent with Table 5-2, aside from uncertainty in passive pipe break frequency which will be treated as a random basic event uncertainty. In accordance with NEI 00-04, the results of the sensitivity study are given to the IDP for consideration in the final risk characterization for components initially classified as LSS that may be reclassified to HSS. The NRC staff finds that the licensee will perform a sensitivity study consistent with Table 5-2 of the NEI 00-04 guidance to address the identified key assumptions and sources of uncertainty in the context of the decisionmaking under consideration for the categorization of the SSC at the time of the risk analysis being performed.

The NRC staff recognizes that the licensee will perform routine PRA changes and updates to assure the PRA continually reflects the as-built, as-operated plant, in addition to changes made to the PRA to support the context of the analysis being performed (i.e., sensitivities). Paragraph 50.69(e) and (f) stipulates the process for feedback and adjustment to assure configuration control is maintained for these routine changes and updates to the PRA(s).

Attachment 6 of the LAR (Reference 1) identifies the SOKC as a source of uncertainty. In RAI 05.d (Reference 3), the NRC staff requested TVA to describe the method used to address the SOKC uncertainty and provide justification to confirm that the method applied is consistent with Appendix 6-A of NUREG-1855, Revision 1 (Revision 10). In response to RAI 05.d (Reference 2), the licensee stated that SOKC is not applied and provided implementation item No. 13 to incorporate SOKC into the MOR consistent with NUREG-1855, Revision 1, prior to using the PRA model to support categorization of SSCs under 10 CFR 50.69. Thus, the NRC staff finds that upon incorporation of the SOKC into the MOR prior to the categorization of SSCs, the SOKC source of uncertainty will be addressed consistent with the guidance provided in NUREG-1855, Revision 1.

3.5.1.7 <u>Summary of IEPRA Acceptability</u>

The NRC staff finds the licensee provided the required information, and the IEPRA (includes internal floods), upon completion of the implementation items provided in Attachment 1 of the supplemental letter dated March 21, 2019 (Reference 2), prior to SSC categorization, is acceptable and therefore meets the requirements set forth in paragraphs 50.69(c)(1)(i) and (ii) of 10 CFR 50.69.

3.5.2 Importance Measures and Integrated Importance Measures

Paragraph 50.69(c)(1)(i) of 10 CFR states, in part, "[c]onsider results and insights from the plant-specific PRA. These requirements are met, in part, by using importance measures and sensitivity studies consistent with the ASME/ANS RA-Sa-2009 PRA standard (Reference 8), as endorsed in RG 1.200, Revision 2 (Reference 7). RG 1.200, Revision 2, states, in part:

> Methods such as importance measure calculations (e.g., Fussell-Vesely Importance [F-V], risk achievement worth [RAW], risk reduction worth [RRW], and Birnbaum Importance) are used to identify the contributions of various events to the estimation of CDF for both individual sequences and the total CDF [i.e., both contributors to the total CDF, including the contribution from the different hazard groups and different operating modes (i.e., full-and low-power and shutdown) and contributors to each contributing sequence are identified].
>
> The results of the Level 2 PRA are examined to identify the contributors (e.g., containment failure mode, physical phenomena) to the model estimation of LERF or LRF [Large Release Frequency] for both individual sequences and the model as a whole [...].

NEI 00-04, Revision 0 (Reference 5), provides guidance where the F-V and RAW importance measures are obtained for each component and each PRA modeled hazard (i.e., separately for the IEPRA (including internal flood) and fire PRA) and the values are then compared to specified criteria as follows:

> Components which have importance measures values that exceed the risk criteria (i.e., F-V greater than 0.005, RAW greater than 2, CCF [common cause failure] RAW greater than 20) are assigned candidate[10] safety-significant.

Section 5.1 of NEI 00-04, Revision 0, recommends that a truncation level of five orders of magnitude below the baseline CDF (or LERF) value should be used for calculating the F-V risk importance measures. The guidance also recommends that the truncation level used should be sufficient to identify all functions with a RAW value greater than 2.

3.5.2.1 Importance Measures

In Section 3.1.1 of the LAR (Reference 1), the licensee stated, in part, "[t]he IDP will be trained in the specific technical aspects and requirements related to the categorization process. Training will address...the interpretation of risk importance measures..." In Section 3.2.3 of the LAR, the licensee stated that because the seismic risk is assessed via an SMA, which is a screening tool, importance measures are not used to determine safety significance. A more detailed review of the characterization for seismic risk using a non-PRA method is discussed in Section 3.5.3.2 of this SE. Similarly, the assessment for the internal fires does not include the

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[10] The term *preliminary* is used synonymous with the term *candidate* in NEI 00-04, Revision 0, guidance. The *candidate* safety significance is not the assigned RISC categorization for the SSC until the IDP has completed its review and approval, consistent with NEI 00-04, Revision 0, guidance, as endorsed by RG 1.201.

generation of importance measures. A more detailed NRC staff review of the use of the FSSEL is provided in Section 3.5.3.1.

The NRC staff finds that the licensee's use and treatment of importance measures are consistent with the guidance in NEI 00-04, Revision 0 (Reference 5), as endorsed in RG 1.201, Revision 1 (Reference 4).

3.5.2.2 Integrated Importance Measures

Section 5.6 of NEI 00-04, Revision 0 (Reference 5), titled, "Integral Assessment," discusses the need for an integrated computation using the available importance measures. It further states, in part, that the "integrated importance measure essentially weights the importance from each risk contributor (e.g., internal events, fire, and seismic PRA models) by the fraction of the total core damage frequency [or large early release frequency] contributed by that contributor." The guidance also provides formulas to compute the integrated F-V, and integrated RAW.

The scope of modeled hazards for TVA includes the IEPRA (includes internal floods). The NRC staff finds that the licensee's use and treatment of importance measures is consistent with the guidance in NEI 00-04, Revision 0, as endorsed in RG 1.201, Revision 1 (Reference 4). A more detailed NRC staff review of the alternate methods for assessing the risk for fire, seismic, and other external hazards is provided in Section 3.5.3 of this SE.

3.5.3 Evaluation of the Use of Non-PRA Methods in SSC Categorization

As required by 10 CFR 50.69(c)(1)(ii), SSC functional importance must be determined using an integrated, systematic process for addressing initiating events, SSCs, and plant operating modes, including those not modeled in the plant-specific PRA. The functions to be identified and considered include design bases functions and functions credited for mitigation and prevention of severe accidents.

As described in Sections 3.2.2, 3.2.3, 3.2.4, and 3.2.5 of the LAR (Reference 1), the licensee's categorization process uses the following non-PRA methods, respectively:

- FSSEL performed for the Fire Protection Program
- SMA performed for the IPEEE;
- Screening analysis performed for the IPEEE for external hazards (e.g., high winds, external flood);
- Screening analysis performed for the IPEEE for other hazards;
- Safe Shutdown Risk Management program consistent with NUMARC 91-06 (Reference 6).

The NRC staff's review of these methods is discussed below.

3.5.3.1 Internal Fire Hazard

In the absence of a FIVE analysis or fire PRA (FPRA) specified in the NEI 00-04 (Reference 5) guidance as approaches to address risk from fire, TVA described in Section 3.2.2 of the LAR (Reference 1) an alternate approach. The alternate approach considers the use of the Sequoyah FSSEL for the evaluation of an SSC's safety significance as it pertains to internal fire events during its 10 CFR 50.69 categorization process. The licensee proposed that all the SSCs on the FSSEL will be categorized as HSS and will not be allowed to be re-categorized by the IDP, consistent with the guidance in NEI 00-04 for non-PRA methods.

The licensee's SSEL is the result of TVA's safe shutdown analysis methodology used to identify, select, and analyze systems, components, and cables needed to demonstrate compliance with 10 CFR Part 50, Appendix R, "Fire Protection Program for Nuclear Power Facilities Operating Prior to January 1, 1979." Although Sequoyah was licensed to operate after January 1, 1979, TVA initially agreed to a license condition to identify and justify differences between its existing and proposed fire protection features and those specified in 10 CFR Part 50, Appendix R, Sections III.G., III.J, III.L, and III.O. Sequoyah's current license condition for fire protection mandates compliance with the approved fire protection program referenced in several documents including NUREG-1232, Volume 2, "Safety Evaluation Report on Tennessee Valley Authority: Sequoyah Nuclear Performance Plan," May 1988 (Reference 19).

The safe shutdown functions necessary to satisfy the performance goals and safe shutdown functions of Appendix R are the reactivity control function, the reactor coolant makeup function, the reactor coolant pressure control function, the decay heat removal function, the process monitoring function, and the support function. TVA used various analytical approaches to ensure that sufficient plant systems are available to perform fire safe shutdown functions. Numerous plant systems are available, alone and in combination with other systems, to provide the required functions and TVA identified a minimum set of plant systems and components to demonstrate that the plant can achieve and maintain safe shutdown. The safe shutdown analysis methodology ensures that the safe shutdown systems selected are capable of: achieving and maintaining subcritical conditions in the reactor; maintaining reactor coolant inventory; achieving and maintaining hot shutdown conditions for an extended period of time; performing cold shutdown repairs needed to achieve and maintain cold shutdown (or, for control building fires that require shutdown from outside of the main control room, achieving cold shutdown conditions within 72 hours); and maintaining cold shutdown conditions thereafter.

Appendix R, Section III.G.1 of 10 CFR Part 50 requires that fire protection features be provided for those SSCs important to safe shutdown and that these features must be capable of limiting fire damage so that one train of systems necessary to achieve and maintain hot shutdown conditions from either the Control Room or the Emergency Control Station(s) is free of fire damage, and that systems necessary to achieve and maintain cold shutdown from either the Control Room or the Emergency Control Station(s) can be repaired within 72 hours.

Appendix R, Section III.G.2 of 10 CFR Part 50 requires that, except as provided in Section III.G.3, where cables or equipment, including associated non-safety circuits that could prevent operation or cause maloperation due to hot shorts, open circuits, or shorts to ground, of redundant trains of systems necessary to achieve and maintain hot shutdown conditions are located within the same fire area outside of primary containment, a means of ensuring that one of the redundant trains is free of fire damage shall be provided.

Appendix R, Section III.G.3 of 10 CFR Part 50 describes where alternative or dedicated shutdown capability and its associated circuits, independent of cables, systems or components in the area, room, zone under consideration should be provided.

In RAI 07.a (Reference 3), the NRC staff requested TVA provide justification that the Fire SSEL method is technically adequate relative to the acceptable methods in NEI 00-04. In response to RAI 07.a (Reference 2), the licensee cited the results of an NEI and industry study of several plants that compared the number of HSS SSCs identified by the approaches and methods used to assess risk for fire: (1) FPRA, (2) FIVE, and (3) FSSEL. TVA stated, "[t]he study concludes that the proposed FSSEL approach is conservative by introducing significantly more SSCs assigned a HSS classification than use of a FPRA or FIVE, [...in addition,] the SSEL approach included all the SSCs identified by the FPRA and the FIVE." While the NRC staff did not review the industry cited study, the NRC staff agrees that the Sequoyah Fire Protection Program identifies a comprehensive list of SSCs (i.e., SSEL) credited to achieve and maintain safe shutdown consistent with the Sequoyah RFOL for applicable requirements to 10 CFR 50.48. The NRC staff finds that the identified SSCs included on the FSSEL and assigned HSS are acceptable for use in the SSC categorization process because the licensee uses the deterministic criteria from Appendix R to identify the functions necessary to achieve and maintain safe shutdown and assigns HSS to all those SSCs that support the Appendix R functions. SSCs that are assigned HSS do not receive relaxation or special treatment under the 10 CFR 50.69 categorization process; therefore, the licensee's approach to use the FSSEL in the 10 CFR 50.69 program is adequate for the categorization of SSCs. The NRC staff finds that TVA's process used to identify, select, and analyze systems, components, and cables to demonstrate compliance with 10 CFR Part 50, Appendix R, will continue to be managed consistent with the applicable regulatory Appendix R requirements.

In RAI 07.d (Reference 3), the NRC staff requested the licensee discuss how the credit for operator actions is considered in the analysis for determining SSCs identified on the FSSEL. In response to RAI 07.d (Reference 2), the licensee stated, "the SSEL Fire Protection Program is based on credible methods (including operator actions) for safely shutting down the plant and maintaining in safe-shutdown for a period of time." In further response to RAI 07.d (Reference 2), the licensee stated "[o]perator actions are not explicitly considered in the safety classification; therefore, there is no assignment of operator action failure probabilities." In Section 3.2.2 of the LAR (Reference 1), TVA stated, "using the Fire SSEL would identify all credited equipment as HSS regardless of their fire damage susceptibility or frequency of challenge." The NRC staff concludes that the licensee's response to the RAI is acceptable because the licensee's use of the Fire SSEL identifies all credited equipment as HSS regardless of their fire damage susceptibility or frequency of challenge.

LAR Figure 3.1 (Reference 1) illustrates a process flowchart that will be used to assess the fire risk during the 10 CFR 50.69 categorization process. The flowchart assesses (1) whether the SSC is on the FSSEL and (2) for SSCs not on the SSEL, whether the SSC is relied upon to maintain safe shutdown for fire. Answers of yes for either of these two questions result in HSS categorization of the SSC. In Section 3.2.2 of the LAR, TVA stated that the identified SSCs pertaining to the regulatory deviations, multiple circuit failures, and additional equipment that is determined to be relied upon to establish and maintain safe shutdown will be retained as HSS for the 10 CFR 50.69 program at Sequoyah. In RAI 07.b(i) (Reference 3), the NRC staff requested the licensee provide clarification regarding the additional SSCs that are not on the FSSEL but will be identified as HSS. In response to RAI 07.b(i) (Reference 2), TVA stated, in part, "those SSCs relied upon to maintain safe shutdown for fire, the reliance referred to in this diamond include SSCs credited by the Fire Protection Program to mitigate multiple spurious operations (MSOs), SSCs credited for exemptions or deviations taken by the Fire Protection Program, and fire protection equipment SSCs (including fire dampers)." In response to RAI 07.c (Reference 2), the licensee further confirmed that the fire protection system SSCs, including fire detection equipment, suppression equipment, and fire dampers are not specifically included in the FSSEL, but that for the purposes of the 10 CFR 50.69 categorization, the fire protection system SSCs are included within the scope of components assigned HSS classification for internal fire hazards.

Additionally, in response to RAI 07.b.(ii) (Reference 2), TVA confirmed, "SSCs assigned as candidate HSS for non-modeled hazards (including) fire are not allowed to be re-categorized to LSS by the IDP; therefore, they remain HSS." In Section 3.2.2 of the LAR (Reference 1), TVA considered the regulatory exemptions related to Fire Safe Shutdown, previously identified fire-induced MSOs, and additional equipment that is relied upon to establish and maintain safe shutdown. In letters dated May 29, 1986, and October 6, 1986, the NRC staff approved deviations from Appendix R of 10 CFR Part 50 for Sequoyah Nuclear Plant Units 1 and 2 (References 23 and 24). In a public meeting with the NRC staff held September 5, 2012, TVA identified SSCs associated with multiple circuit failure faults and provided planned modifications to resolve the undesired operation of SSCs for Sequoyah Units 1 and 2 (Reference 25). Based on the RAI supplement, the NRC staff finds that retaining the SSCs identified on the FSSEL as HSS is conservative and acceptable because TVA's approach to using the FSSEL considers additional equipment that includes regulatory exemptions related to the fire safe shutdown program, previously identified fire-induced MSOs, and additional equipment that is relied upon to establish and maintain safe shutdown that will be retained as HSS.

Section 9.2.3 of NEI 00-04 (Reference 5) guidance for non-safety-related SSCs identified as candidate LSS states, in part, for SSCs, which are important-to-safety, the IDP must consider if the risk information used in the categorization process provides an adequate basis for categorizing the SSC as RISC-4. The FSSEL is a screening approach; therefore, there are no importance measures used in determining safety significance related to the fire hazard and assessment for LSS SSCs can be limited. Regulatory Guide 1.201 (Reference 4) states, in part, "the implementation of all processes described in NEI 00-04 (i.e., Sections 2 through 12) is integral to provide reasonable confidence" and that "all aspects of NEI 00-04 must be followed to achieve reasonable confidence in the evaluations required by §50.69(c)(1)(iv)." For SSCs identified as safety-related candidate LSS (i.e., RISC-3), Section 9.2.2 of the NEI 00-04 guidance stipulates consideration by the IDP for DID and safety margin implications for confirming LSS. The NRC staff finds that, in the absence of importance measures to assess the safety significance related to the fire hazard, the other assessments (e.g., DID, safety margin, etc.) for determining the risk categorization of SSCs not included on the Fire SSEL will be performed consistent with the NEI 00-04 guidance, therefore assuring the SSC has been appropriately assigned candidate LSS.

In Section 3.2.2 of the LAR (Reference 1), TVA stated, in part, "the Fire SSEL and the identification of additional equipment relied upon to establish and maintain safe shutdown reflects the current as-built, as-operated plant and that changes to the plant will be evaluated to determine their impact to the equipment list and the categorization process." The FSSEL is an active document that supports the Sequoyah Appendix R program. Changes are managed within the Fire Protection Program consistent with the Sequoyah License Condition 2.C.16 and 2.C.13 for Units 1 and 2, respectively. Therefore, the NRC staff finds that future changes to the Sequoyah Appendix R SSEL will be evaluated to determine their impact on the FSSEL and risk categorization process.

Based upon the NRC staff's review of TVA's approach to using the FSSEL provided in Section 3.2.2 of the LAR (Reference 1) and supplemented in response to RAI 07 (Reference 2), the NRC staff finds the licensee's approach to use the Sequoyah FSSEL to assess the risk for internal fires, when integrated with the other steps/elements provided in the NEI 00-04 guidance as endorsed by the NRC, is acceptable for use in the 10 CFR 50.69 SSC categorization program.

3.5.3.2 Seismic Risk

NEI 00-04, Revision 0 (Reference 5), as endorsed in RG 1.201, Revision 1 (Reference 4), states, in part:

> In the event an SMA is used, the categorization process is, once again, more conservative (i.e., designed to identify more SSCs as safety-significant). This is due to the fact that the SMA analysis is a screening tool. As a screening tool, importance measures are not available to identify safety significance. The NEI 00-04 approach identifies all system functions and associated SSCs that are involved in the seismic margin success paths as safety-significant. This measure of safety significance assures that the SSCs that were required to maintain low seismic risk are retained as safety-significant.

In Section 3.2.3 of the LAR (Reference 1), the licensee proposed using the SMA performed for the IPEEE in response to Generic Letter 88-20, Supplement 4 (Reference 26) that was based on the EPRI SMA method described in EPRI NP-6041-SL, Revision 1, "A Methodology for Assessment of Nuclear Power Plant Seismic Margin," November 2017 (Reference 22). The SMA method includes the development of the seismic SSEL which contains the components that would be needed during and after a seismic event. The seismic SSEL identifies one preferred and one alternate path capable of achieving and maintaining safe shutdown conditions for at least 72 hours following an earthquake.

In Section 3.2.3 of the LAR, the licensee stated that "[a]n evaluation was performed of the as-built, as-operated plant against the SMA SSEL." The licensee discussed that the evaluation considered potential impacts to equipment credited on the seismic SSEL and appropriate changes were made, including documentation to the credited equipment. The licensee further stated in the LAR that "future changes to the plant will be evaluated to determine their impact on the SMA and risk categorization process." The NRC staff finds the licensee's use of the SMA to assess seismic risk acceptable and consistent with NEI 00-04, Revision 0, as endorsed in RG 1.201, Revision 1, and therefore meets the requirements set forth in 10 CFR 50.69(c)(1)(ii).

3.5.3.3 External Hazards and Other Hazards (Non-Seismic)

External hazards were initially evaluated by the licensee during the IPEEE (Reference 27). This hazard category includes all non-seismic external hazards such as high winds, external floods, transportation and nearby facility accidents, and other hazards. In the safety evaluation report for the Sequoyah IPEEE for Units 1 and 2, the staff states, in part, "[t]he high winds, floods, transportation and other external events (HFO) areas were eliminated based on either compliance with 1975 NRC Standard Review Plan (SRP) criteria or on the basis of a bounding probabilistic assessment resulting in a CDF estimate less than 10^{-6} per reactor year, i.e., below the NUREG-1407 screening criterion" (References 11 and 21).

In Section 3.2.4 of the LAR (Reference 1), the licensee stated, in part, all other external hazards (i.e., not seismic or fire hazards) were screened from applicability to SQN [Sequoyah Nuclear Generating Plant] per a plant-specific evaluation in accordance with GL 88-20 and updated to use the criteria in ASME/ANS PRA Standard RA-Sa-2009. In RAI 02.a (Reference 3), the NRC staff requested TVA identify the external hazards that will be evaluated according to the flow chart in NEI 00-04, Section 5.4, Figure 5-6 (Reference 5). In response to RAI 02.a (Reference 2), the licensee stated that TVA will subject the external hazards (excluding internal fires and seismic hazards) to the process described by the flow chart in NEI 00-04, Figure 5-6. NEI 00-04, Figure 5-6 provides guidance to be used to determine SSC safety significance for other external hazards (excluding internal fires and seismic hazards). The NRC staff finds that TVA will assess the other external hazards consistent with Figure 5-6 of NEI 00-04 as endorsed in RG 1.201, Revision 1.

The guidance in NEI 00-04, Revision 0, states, in part,

> If it can be shown that the component either did not participate in any screened scenarios or, even if credit for the component was removed, the screened scenario would not become unscreened, then it is considered a candidate for the LSS category.

In Attachment 4 of the LAR (Reference 1), the licensee provided the results of the plant-specific evaluation that assessed the IPEEE results to the updated endorsed criteria in the ASME/ANS RA-Sa-2009 PRA Standard. The NRC notes, this plant-specific evaluation or its results were not peer reviewed against Part 6 of the ASME/ANS Ra-SA-2009 PRA Standard (Reference 8) as endorsed in RG 1.200, Revision 2. Therefore, in RAI 02.a(i)-(iii) (Reference 3), the NRC staff requested the licensee provide detailed justification for screening external hazards (i.e., external flood, high winds, and tornados) using the criteria in Part 6 of the ASME/ANS RA-Sa-2009 PRA Standard). In response to RAI 02.a.(i) (Reference 2), the licensee stated, "[i]n Attachment 4 of the LAR, TVA screened the external flooding hazard using criterion PS1 and PS4. Extreme winds and tornado hazards were screened using criterion PS1 and PS3. Based on further evaluation, TVA has determined that the more appropriate screening criteria should be C5 for external flooding and PS2 for extreme winds and tornados [...]." In RAI 02.a(iii) (Reference 3), the licensee provided further justification to support the determination for applying PS2 for the extreme winds and tornado hazards. To confirm Sequoyah meets the 1975 SRP, TVA reviewed the design bases against the SRP requirements, in addition to any changes made to the plant subsequent to the design analysis. In conclusion, TVA stated, "[f]or other external events, it was found that no vulnerabilities exist outside of the screening thresholds in the SRP, [t]herefore the SQN design meets the 1975 SRP criteria for extreme winds and tornadoes [...]."
Section 6-2.3 of the ASME/ANS Ra-Sa-2009 PRA Standard (Reference 8) for criterion (a) states, in part, that an event can be screened out if it meets the criteria in the NRC's 1975 SRP or a later revision. The licensee reviewed the design basis against the SRP requirements and reviewed subsequent changes; therefore, NRC staff finds that it is acceptable to assess extreme winds and tornadoes consistent with Figure 5-6 of NEI 00-04 guidance as endorsed by RG 1.201, Revision 1.

For RAI 02.a.(ii) (Reference 3), the NRC staff requested the licensee provide justification for concluding that the screening criterion PS4 for the external flooding hazard confirms the CDF is less than 1E-06 per reactor-year. In response to RAI 02.a.(ii) (Reference 2), TVA committed to provide the NRC with a revised SQN warning time analysis. In Attachment 1 of Enclosure 1 of the letter dated March 21, 2019 (Reference 2), the licensee provided an implementation item that includes: "[w]ith respect to the external flooding hazards, TVA shall re-confirm that there is sufficient time to eliminate the source of the threat or to provide an adequate response in accordance with screening criterion C5, prior to 50.69 categorization." In response to RAI 03 (Reference 2), TVA provided a license condition that states, "[p]rior to implementation of the provisions of 10 CFR 50.69, TVA shall complete the items listed in Enclosure 1, Attachment 1, "SQN 10 CFR 50.69 PRA Implementation Items," in TVA letter CNL-19-002 [...]." The NRC staff finds that upon confirmation that there is sufficient time to eliminate the source of the threat or response in accordance with the screening criterion C5 in Part 6 of the ASME/ANS Ra-SA-2009 PRA standard as endorsed by the NRC, it is acceptable to assess the external flooding consistent with Figure 5-6 of NEI 00-04 as endorsed by the NRC.

In RAI 02.b.i (Reference 3), the NRC staff requested that the licensee identify and justify what type of SSCs, if any, are credited in the screening of external hazard(s), including passive, active, and temporary features. In response to RAI 02.b(i) (Reference 2), the licensee stated "[t]he SQN IPEEE concluded all Other External Hazards (excluding seismic and internal fires) were screened from further consideration in the examination of these hazards. There was no identification of SSCs that supported the screening of these hazards."

In summary, the use of the Sequoyah IPEEE results described by the licensee in the LAR (Reference 1), supplemental information provided in response to RAI 02 (Reference 2), and the licensee's assessment of the other external hazards (i.e., high winds, tornadoes, and external flood) is consistent with Section 5 of NEI 00-04, Revision 0, as endorsed in RG 1.201, Revision 1. The NRC staff concludes that the licensee's treatment of other external hazards is acceptable and meets 10 CFR 50.69(c)(1)(ii).

3.5.3.4 Shutdown Risk

Consistent with the guidance in NEI 00-04, Revision 0 (Reference 5), the licensee proposed using the shutdown safety assessment based on NUMARC 91-06 (Reference 6). NUMARC 91-06 provides considerations for maintaining DID for the five key safety functions during shutdown, namely, decay heat removal capability, inventory control, power availability, reactivity control, and containment-primary/secondary. NUMARC 91-06 also specifies that a DID approach should be used with respect to each defined shutdown key safety function. This is accomplished by designating a running and an alternative system/train to accomplish the given key safety function.

The use of NUMARC 91-06 (Reference 6) described by the licensee in the submittal is consistent with the guidance in NEI 00-04, Revision 0 (Reference 5), as endorsed in the NRC in RG 1.201, Revision 1 (Reference 4). The approach uses an integrated and systematic process to identify HSS components, consistent with the shutdown evaluation process. Therefore, the NRC finds that the licensee's use of NUMARC 91-06 is acceptable, and meets the requirements set forth in 10 CFR 50.69(c)(1)(ii).

3.5.3.5 Component Safety Significance Assessment for Passive Components

Passive components are not modeled in the PRA; therefore, a different assessment method is necessary to assess the safety significance of these components. Passive components are those components having only a pressure retaining function. This process also addresses the passive function of active components such as the pressure/liquid retention of the body of a motor-operated valve.

In Section 3.1.2 of the LAR (Reference 1), the licensee proposed using a categorization method for passive components not cited in NEI 00-04, Revision 0 (Reference 5), or RG 1.201, Revision 1 (Reference 4), for passive component categorization, but was approved by the NRC for Arkansas Nuclear One, Unit 2 (ANO-2) (Reference 12). The ANO-2 methodology is a risk-informed safety classification and treatment program for repair/replacement activities for Class 2 and 3 pressure retaining items and their associated supports (exclusive of Class CC and MC items), using a modification of the ASME Code Case N-660, "Risk-Informed Safety Classification for Use in Risk-Informed Repair/Replacement Activities, Section XI, Division 1" (Reference 20). The ANO-2 methodology relies on the conditional core damage and large early release probabilities associated with pipe ruptures. Safety significance is generally measured by the frequency and the consequence of, in this case, pipe ruptures. Treatment requirements (including repair/replacement) only affect the frequency of passive component failure. Categorizing solely based on consequences, which measures the safety significance of the pipe given that it ruptures, is conservative compared to including the rupture frequency in the categorization. The categorization will not be affected by changes in frequency arising from changes to the treatment. Therefore, the NRC staff finds that the use of the repair/replacement methodology is acceptable and appropriate for passive component categorization of Class 2 and Class 3 SSCs.

In Section 3.1.2 of the LAR (Reference 1), the licensee stated, "[t]he passive categorization process is intended to apply the same risk-informed process accepted in the ANO2-R&R-004 for the passive categorization of Class 2, 3, and non-class components. Consistent with ANO2-R&R-004, Class 1 pressure retaining SSCs in the scope of the system being categorized will be assigned HSS and cannot be changed by the IDP." The NRC staff finds the licensee's proposed approach for passive categorization is acceptable for the 10 CFR 50.69 SSC categorization process.

3.5.3.6 Maintain Defense-in-Depth (NEI 00-04, Section 6)

Section 6 of NEI 00-04, Revision 0 (Reference 5), provides guidance on assessment of DID. Figure 6-1 in NEI 00-04, Revision 0, provides guidance to assess design basis DID based on the likelihood of the design-basis internal initiating event and the number of redundant and diverse trains nominally available to mitigate the initiating event. The likelihoods of the initiating events are binned. The bins for the different likelihoods consider the number of mitigating trains nominally available and if HSS is inadvertently assigned for SSCs that require a fewer number of mitigating trains. Section 6 of NEI 00-04, Revision 0, also provides guidance to assess containment DID based on preserving containment isolation and long-term containment integrity and on preventing containment bypass and early hydrogen burns.

RG 1.201, Revision 1 (Reference 4), endorses the guidance in Section 6 of NEI 00-04 (Reference 5), but notes that the containment isolation criteria in this section of the guidance are separate and distinct from those set forth in 10 CFR 50.69(b)(1)(x). The criteria in 10 CFR 50.69(b)(1)(x) are to be used in determining which containment penetrations and valves may be exempted from the Type B and Type C leakage testing requirements in both Options A and B of Appendix J, "Primary Reactor Containment Leakage Testing for Water-Cooled Power Reactors," to 10 CFR Part 50. The criteria provided in paragraph 50.69(b)(1)(x) of 10 CFR are not to determine the proper RISC category for containment isolation valves or penetrations.

In Section 3.1.1 of the LAR (Reference 1), the licensee clarified that it will require an SSC to be categorized as HSS based on the DID assessment performed in accordance with NEI 00-04, Revision 0. The NRC staff finds that the licensee's process is consistent with the

NRC-endorsed guidance in NEI 00-04 and therefore fulfills the 10 CFR 50.69(c)(1)(iii) criterion that requires DID to be maintained.

3.5.4 Preliminary Engineering Categorization of Functions (NEI 00-04, Section 7)

All the information collected and evaluated in the licensee's engineering evaluations is provided to the IDP as described in Section 7 of NEI 00-04, Revision 0 (Reference 5). The IDP will make the final decision about the safety significance of SSCs based on guidelines in NEI 00-04, Revision 0, the information they receive, and their expertise.

In Section 3.1.1 of the LAR (Reference 1), the licensee stated, in part, "...if any SSC is identified as HSS from either the integrated PRA component safety significance assessment (Section 5 of NEI 00-04) or the DID assessment (Section 6), the associated system function(s) would be identified as HSS." The licensee also stated that, "[o]nce a system function is identified as HSS, then all the components that support that function are preliminary HSS."

The NRC staff finds that the above description provided by the licensee for the preliminary categorization of functions is consistent with NEI 00-04, Revision 0 (Reference 5), as endorsed in RG 1.201, Revision 1 (Reference 4), and is therefore acceptable.

3.5.5 Risk Sensitivity Study (NEI 00-04, Section 8)

Paragraph 50.69(c)(1)(iv) of 10 CFR requires, in part, that any potential increases in CDF and LERF resulting from changes to treatment are small. The categorization process described in NEI 00-04, Revision 0, includes an overall risk sensitivity study for all the LSS components to assure that if the unreliability of the components was increased, the increase in risk would be small (i.e., meet the acceptance guidelines of RG 1.174, Revision 3 (Reference 9)).

Section 3.1.1 of the LAR (Reference 1) states that an unreliability factor of 3 will be used for the sensitivity studies described in Section 8, "Risk Sensitivity Study," of NEI 00-04, Revision 0 (Reference 5). Section 3.2.7 of the LAR further confirms that a cumulative sensitivity study will be performed where the failure probabilities (unreliability and unavailability, as appropriate) of all LSS components modeled in PRAs for all systems that have been categorized are increased by a factor of 3. The NRC staff finds the application of a factor of 3 for the sensitivities is consistent with the guidance in NEI 00-04, Revision 0, as endorsed by RG 1.201, Revision 1 (References 5 and 4).

In Section 3.1.1 of the LAR (Reference 1), for the "Overall Categorization Process," TVA specifically noted that "the implementation of all processes described in NEI 00-04 (i.e., Sections 2 through 12) is integral to providing reasonable confidence" and that "all aspects of NEI 00-04 must be followed to achieve reasonable confidence in the evaluations required by §50.69(c)(1)(iv)." This sensitivity study together with the periodic review process discussed in Section 3.6 of this SE, assure that the potential cumulative risk increase from the categorization is maintained acceptably low. The performance monitoring process monitors the component performance to ensure that potential increases in failure rates of categorized components are detected and addressed before reaching the rate assumed in the sensitivity study. The NRC staff finds that the licensee will perform the risk sensitivity study consistent with the guidance in Section 8 of NEI 00-04, Revision 0 (Reference 5), and, therefore, will assure that the potential cumulative risk increase from the categorization is maintained acceptably low, as required by 10 CFR 50.69(c)(1)(iv).

3.5.6 IDP Review and Approval (NEI 00-04, Revision 0, Sections 9 and 10)

As required by 10 CFR 50.69(c)(2), the SSCs must be categorized by an IDP staffed with expert, plant knowledgeable members whose expertise includes, at a minimum, PRA, safety analysis, plant operations, design engineering, and system engineering. In Section 3.1.1 of the LAR (Reference 1), the licensee stated that the IDP will be composed of a group of at least five experts who collectively have expertise in plant operation, design (mechanical and electrical) engineering, system engineering, safety analysis, and PRA. Therefore, the IDP will comprise the required expertise.

The guidance in NEI 00-04, Revision 0 (Reference 5), as endorsed in RG 1.201, Revision 1 (Reference 4), provides confidence that the IDP expertise is sufficient to perform the categorization and that the results of the different evaluations (PRA and non-PRA) are used in an integrated, systematic process as required by 10 CFR 50.69(c)(1)(ii). In Section 3.1.1 of the LAR (Reference 1), the licensee discussed that at least three members of the IDP will have a minimum of five years of experience at the plant, and there will be at least one member of the IDP who has a minimum of three years of experience in modeling and updating of the plant-specific PRA. The licensee further states that the IDP will be trained in the specific technical aspects and requirements related to the categorization process. This training will address, at a minimum, the purpose of the categorization; present treatment requirements for SSCs including requirements for design basis events; PRA fundamentals; details of the plant specific PRA including the modeling, scope, and assumptions, the interpretation of risk importance measures, and the role of sensitivity studies and the change-in-risk evaluations; and the DID philosophy and requirements to maintain this philosophy. In Attachment 1 of Enclosure 1 to a letter dated March 21, 2019 (Reference 2), the licensee confirmed as a prerequisite for categorization that Sequoyah will establish procedure(s) prior to the use of the categorization process on a plant system. The licensee also confirmed the procedure(s) will specifically include an element for the IDP member qualification requirements. The NRC staff finds that the licensee's IDP areas of expertise meet the requirements in 10 CFR 50.69(c)(2) and the additional descriptions of the IDP characteristics, training, processes, and decision guidelines are consistent with NEI 00-04, Revision 0 (Reference 5), as endorsed in RG 1.201, Revision 1 (Reference 4).

Section 9.2.2, "Review of Safety Related Low Safety-Significant Functions/SSCs," of NEI 00-04, Revision 0 (Reference 5), which is performed by the IDP, states, in part, "in making their assessment, the IDP should consider the impact of loss of the function/SSC against the remaining capability to perform the basic safety functions [...]." This section also provides seven specific questions that should be considered by the IDP for making the final determination of the safety-significance for each function/SSC.

The IDP's authority to change component categorization from preliminary HSS to LSS is limited. Consistent with the guidance in NEI 00-04, Revision 0 (Reference 5), and Table 3-1 provided by the licensee in the LAR (Reference 1), components found to be HSS from the following aspects of the process cannot be re-categorized by the IDP: internal events PRA, non-PRA approaches (i.e., FSSEL, SMA, shutdown events, other hazards, external events (includes high winds), DID, and passive categorization. SSCs identified as HSS through sensitivity studies outlined in Section 5 of NEI 00-04, may be presented to the IDP for categorization as LSS, if this determination is supported by the integrated assessment process and other elements of the categorization process.

In RAI 06 (Reference 3), the NRC staff requested the licensee clarify how the IDP will collectively assess the seven questions in Section 9.2.2 of NEI 00-04 to identify a function/SSC as LSS as opposed to HSS. In response to RAI 06.a (Reference 2), the licensee confirmed that the assessment of the qualitative considerations is agreed upon by the IDP in accordance with Section 9.2 of the NEI 00-04 guidance. The licensee further stated, in part, "TVA procedures governing the IDP will require that if any one of the seven statements for consideration has a 'FALSE' response, the function risk will be assigned a classification of HSS. If all seven responses are 'TRUE', a function risk of LSS will be assigned; however, each 'TRUE' response requires a supporting justification for confirming the basis for the decision.

The IDP is responsible for reviewing the preliminary assessment to the same level of detail as the 10 CFR 50.69 team (i.e., all considerations for all functions are reviewed). The IDP may confirm the preliminary function risk and associated justification or may direct that it be changed based upon their expert knowledge. The qualitative criteria are the direct responsibility of the IDP, as such changes may be made from preliminary HSS to LSS or from preliminary LSS to HSS at the discretion of the IDP. The licensee further confirmed that the final assessment of the seven qualitative questions in Section 9.2 of NEI 00-04 (Reference 5) is the IDP's responsibility and that the final categorization of the function will be HSS when any one of the seven questions cannot be confirmed (false response) for that function. The NRC staff finds this acceptable and consistent with the guidance in NEI 00-04, Revision 0, as endorsed in RG 1.201 (Reference 4).

The IDP may change the categorization of a component from LSS to HSS based on its assessment and decisionmaking. As outlined in Section 10.2, "Detailed SSC Categorization of NEI 00-04, Revision 0 (Reference 5), and confirmed by the licensee in Section 3.1.1 of the LAR, the IDP's ability to re-categorize components supporting an HSS function from HSS to LSS is limited and only available to the IDP based upon the prescribed steps in the NEI 00-04 guidance as endorsed by RG 1.201, Revision 1 (Reference 4). The steps of the process are performed at either the functional level, component level, or both. For the Sequoyah SSC categorization process, the IDP can re-categorize components from HSS to LSS using the qualitative criteria outlined in Section 9.2 of the NEI 00-04 guidance and PRA sensitivities used to assess the results of the IEPRA (includes internal floods).

As discussed in NEI 00-04, Revision 0 (Reference 5), the only LSS SSC requirements that are relaxed for RISC-3 (LSS) SSCs are those related to treatment, not design or capability, and 10 CFR 50.69(d)(2)(i) requires the licensee ensures, with reasonable confidence, that RISC-3 SSCs remain capable of performing their safety-related functions under design basis conditions. Therefore, the NRC staff finds that the IDP for the Sequoyah categorization process is consistent with the endorsed guidance in NEI 00-04, Revision 0, and, therefore, fulfills 10 CFR 50.69(c)(1)(iv).

3.6 Programmatic Configuration Control (NEI 00-04, Revision 0, Sections 11 and 12)

Paragraph 50.69(c)(1)(ii) of 10 CFR requires, in part, that all aspects of the integrated, systematic process used to characterize SSC importance must reasonably reflect the current plant configuration and operating practices and applicable plant and industry operating experience. Sections 11 and 12 of NEI 00-04, Revision 0 (Reference 5), includes discussion on Periodic Review and program documentation and change control. Maintaining change control and periodic review will also maintain confidence that all aspects of the 10 CFR 50.69 program and risk categorization for SSCs continually reflect the Sequoyah as built, as-operated plant. A more detailed staff review is provided as follows:

3.6.1 Periodic Review (NEI 00-04, Revision 0, Section 12)

Section 50.69(e), “Feedback and Process Adjustment,” of 10 CFR requires periodic updates to the licensee's PRA and SSC categorization must be performed. Changes over time to the PRA and to the SSC reliabilities are inevitable and such changes are recognized by the 10 CFR 50.69(e) requirement for periodic updates.

Section 11.2 of NEI 00-04, Revision 0 (Reference 5), titled, “Following Initial Implementation,” states, in part, “[t]he periodic update of the plant PRA may affect the results of the categorization process. If the results are affected, the licensee must make adjustments as necessary to either the categorization or treatment processes to maintain the validity of the processes.” In Section 3.2.6 of the LAR (Reference 1), the licensee described the process for maintaining and updating the Sequoyah PRA models used for the 10 CFR 50.69 categorization process. Consistent with NEI 00-04, the licensee stated, “[t]he TVA risk management process ensures that the applicable PRA mode(s) used in this application continue to reflect the as-built and as-operated plant for each of the SQN units.” The licensee's process includes provisions for: monitoring issues affecting the PRA models (e.g., due to changes in the plant, errors or limitations identified in the model, industry operational experience); assessing the risk impact of unincorporated changes; and controlling the model and associated computer files. The process also includes reevaluating previously categorized systems to ensure the continued validity of the categorization.

Routine PRA updates are performed every two refueling cycles at a minimum. The NRC staff finds the risk management process described by the licensee in the LAR is consistent with Section 12 of NEI 00-04, Revision 0 guidance as endorsed by the NRC.

3.6.2 Program Documentation and Change Control (NEI 00-04, Revision 0, Section 11)

Section 50.69(f) of 10 CFR requires, in part, program documentation, change control, and records. In Section 3.2.6 of the LAR (Reference 1), the licensee stated that it will implement a process that addresses the requirements in Section 11 of NEI 00-04, Revision 0 (Reference 5), pertaining to program documentation and change control records. Section 3.1.1 of the LAR states that the RISC categorization process documentation will include the following ten elements:

- Program procedures used in the categorization
- System functions, identified and categorized with the associated bases
- Mapping of components to support function(s)
- PRA model results, including sensitivity studies
- Hazards analyses, as applicable
- Passive categorization results and bases
- Categorization results including all associated bases and RISC classifications
- Component critical attributes for HSS SSCs
- Results of periodic reviews and SSC performance evaluations
- IDP meeting minutes and qualification/training records for the IDP members

Attachment 1, SQN 10 CFR 50.69 PRA Implementation Items,” of Enclosure 1 to the letter dated March 21, 2019 (Reference 2), provides the following steps/elements that the licensee will include in the 10 CFR 50.69 programmatic procedures prior to the use of the categorization process: (1) IDP member qualification requirements, (2) qualitative assessment of system functions (3) component safety significance assessment (4) assessment of DID and safety margin (5) review by the IDP (6) overall risk sensitivity study; (7) periodic review and, (8) documentation requirements identified in Section 3.1.1 of the LAR (Reference 1).

The NRC staff also recognizes that for facilities licensed under 10 CFR Part 50, Appendix B Criterion VI, for Document Control, procedures are considered formal plant documents that require "[m]easures shall be established to control the issuance of documents, such as instructions, procedures, and drawings, including changes thereto, which prescribe all activities affecting quality." The NRC staff finds that the elements provided in Section 3.1.1 of the LAR, in addition to the list of implementation items provided in Attachment 1 of Enclosure 1 to the letter dated March 21, 2019 (Reference 2), for the Sequoyah 10 CFR 50.69 categorization process will be documented in formal licensee procedures consistent with Section 11 of NEI 00-04, Revision 0 (Reference 5), as endorsed by the NRC in RG 1.201, Revision 1 (Reference 4), and therefore sufficient for meeting the 10 CFR 50.69(f) requirement for program documentation, change control and records.

3.7 Summary of 10 CFR 50.69 Categorization Process

The NRC staff finds the PRAs and use of the non-PRA methods described by the licensee in the submittal (Reference 1) as supplemented in letter dated March 21, 2019 (Reference 2) acceptable for use in the SSC categorization process. The NRC staff approves the use of the following approaches and methods in the licensee's 10 CFR 50.69 categorization process:

- IEPRA (includes internal flood) to assess the risk from internal events and internal flood, respectively;
- Sequoyah Fire SSEL to assess fire risk;
- SMA performed for the Sequoyah IPEEE (Reference 11) to assess seismic risk;
- Screening analysis performed for the IPEEE (Reference 11) to assess the risk for high winds, external floods, and other hazards;
- Shutdown Safety Management Plan consistent with NUMARC 91-06 (Reference 6) to assess shutdown risk;
- ANO-2 passive categorization method to assess passive components for Class 2 and 3 SSCs and their associated supports (Reference 12)

The NRC staff reviewed all of the primary steps outlined in Section 3.2 of this SE used by the licensee in the 10 CFR 50.69 categorization process to assess the safety significance of active and passive components while ensuring the SSC's intended functions remain intact. The NRC staff concludes that the licensee's categorization process adequately implements 10 CFR 50.69 using models, methods, and approaches consistent with NEI 00-04, Revision 0 (Reference 5), as endorsed by the NRC, and therefore, satisfies the requirements of 10 CFR 50.69(c). The NRC staff finds the licensee's proposed categorization process acceptable for categorizing the safety significance of SSCs. Specifically, the NRC staff concludes that the licensee's categorization process:

(1) Considers results and insights from plant-specific internal events PRA that have been subjected to a peer review process against RG 1.200, Revision 2 (Reference 7), as reviewed in Section 3.5.1.1 through 3.5.1.6 of this SE, and with the completion of the implementation items, will be of sufficient quality and level of detail to support the categorization process , and therefore, meets the requirements in 10 CFR 50.69(c)(1)(i);

(2) Determines SSC functional importance using an integrated systematic process that reasonably reflects the current plant configuration, operating practices, and applicable plant and industry operational experience, as reviewed in Sections 3.5 and 3.6, and therefore, meets the requirements in 10 CFR 50.69(c)(1)(ii);

(3) Maintains DID, as reviewed in Section 3.5.3.6 of this SE, and therefore, meets the requirements in 10 CFR 50.69(c)(1)(iii);

(4) Includes evaluations that provide reasonable confidence that for SSCs categorized as RISC-3, sufficient safety margins are maintained and that any potential increases in CDF and LERF resulting from changes in treatment are small, as reviewed in Sections 3.5.3.6 and 3.5.7 of this SE, and therefore, meets the requirements in 10 CFR 50.69(c)(1)(iv);

(5) Is performed for entire systems and structures, rather than for selected components within a system or structure, as reviewed in Section 3.5 of this SE, and therefore, the requirements in 10 CFR 50.69(c)(1)(v) will be met upon implementation; and

(6) Includes categorization by IDP, staffed with expert, plant-knowledgeable members whose expertise includes, at a minimum, PRA, safety analysis, plant operation, design engineering and system engineering, as reviewed in Section 3.5.8 of this SE, and therefore, meets the requirements in 10 CFR 50.69(c)(2).

In Attachment 1 of Enclosure 1 to the letter dated March 21, 2019, the licensee provided a list of implementation items that will establish procedure(s) prior to the use of the categorization process on a plant system. The list of the implementation items encompasses in its entirety the steps/elements described in the NEI 00-04, Revision 1 as endorsed by the NRC and reviewed by the NRC staff in this SE. Therefore, the NRC staff concludes that upon the completion of these implementation items the Sequoyah IEPRA (includes internal floods) and SSC categorization process will be updated and controlled consistent with 10 CFR 50.69(e) and (f) for ensuring that the categorization of SSCs continue to reflect the as-built-as-operated plant design.

The implementation items provided by the licensee in Attachment 1 of Enclosure 1 to the letter dated March 21, 2019 (Reference 2), are as follows:

1. IDP member qualification requirements.

2. Qualitative assessment of system functions. System functions are qualitatively categorized as preliminary HSS or LSS based on the seven criteria in Section 9 of NEI 00-04 (see Section 3.1.1 of the original LAR (CNL-17-010)). Any component supporting an HSS function is categorized as preliminary HSS. Components supporting LSS function are categorized as preliminary LSS.

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

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

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

6. Risk sensitivity study. For PRA-modeled components, an overall risk sensitivity study is used to confirm that the population of preliminary LSS components results in acceptably small increases to CDF and LERF and meets the acceptance guidelines of RG 1.174 (Reference 9).

7. Internal Event Risks: Internal Events including internal flooding PRA model Revision 3, dated August 5, 2014 or a later updated model as described in section 3.2.6 of this enclosure will be used. This model was accepted by NRC in NRC Letter to TVA, "Sequoyah Nuclear Plant, Units 1 and 2 - Issuance of Amendments for the Conversion to the Improved Technical Specifications with Beyond Scope Issues (TAC Nos. MF3128 and MF3129)," dated September 30, 2015 (ML15238B460).

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

9. Documentation requirements per Section 3.1.1 of enclosure 1 of the LAR (CNL-17-010) (ML18075A365).

10. TVA procedures governing the IDP will require that if any one of the seven statements for consideration has a 'FALSE' response the function risk will be assigned a classification of HSS.

11. TVA procedures will require the Categorization Team to consider the seven statements of consideration in addition to the IDP.

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

12. As documented in the F&O Closure Report, all changes initiated by the F&O resolutions were confirmed by the Integrated Assessment Team to have been incorporated into the living model and associated documentation. TVA shall

update the Model of Record (MOR) with this information prior to system categorization.

13. TVA shall re-introduce the SOKC into the MOR prior to using the PRA model to support categorization of SSCs under 10 CFR 50.69.

14. With respect to the external flooding hazards, TVA shall re-confirm that there is sufficient time to eliminate the source of the threat or to provide an adequate response in accordance with screening criterion C5, prior to 50.69 categorization.

3.8 Overall Technical Review Summary and Final Change to the License

Based on the staff's review of the LAR and the licensee's responses to the staff's RAIs, the staff identified specific actions, as described below, that are identified as being necessary to support the NRC staff's conclusion that the proposed program meets the requirements in 10 CFR 50.69 and the guidance in RG 1.201, Revision 1 (Reference 4), and NEI 00-04, Revision 0 (Reference 5). Note: Additional actions (e.g., final procedures and proposed alternative treatment) need not, and have not been developed, submitted, or reviewed by the staff for issuance of the SE, but will be completed before implementation of the program as specified in the 10 CFR 50.69 rule.

The NRC staff's finding on the acceptability of the PRA evaluation in the licensee's proposed 10 CFR 50.69 process is conditioned upon the License Condition provided below that delineates completion of the implementation items and/or list prerequisites to address changes to the PRA model or documentation. These implementation items are identified in Attachment 1 of Enclosure 1 of the letter dated March 21, 2019 (Reference 2). The NRC staff finds the clarifications to the NEI 00-04, Revision 0 guidance (Reference 5) and other changes that were described by the licensee to be routine and systematically addressed through the configuration management and control and periodic update processes as described in Section 3.6 of this SE.

In response to RAI 03 (Reference 2), the licensee proposed the following amendment to the RFOLs for Sequoyah, Units 1 and 2. The proposed license condition states:

> Adoption of 10 CFR 50.69, "Risk-Informed categorization and treatment of structures, systems and components for nuclear power plants"
>
> (1) TVA is approved to implement 10 CFR 50.69 using the processes for categorization of Risk Informed Safety Class (RISC)-1, RISC-2, RISC-3, and RISC-4 structures, systems, and components (SSCs) using: Probabilistic Risk Assessment (PRA) model to evaluate risk associated with internal events, including internal flooding; using the fire safe shutdown equipment list in the SQN Fire Protection Report referenced in the Updated Final Safety Analysis Report to evaluate internal fire events; the NUMARC 91-06 shutdown safety assessment process to assess shutdown risk; the Arkansas Nuclear One, Unit 2 (ANO-2) passive categorization method to assess passive component risk for Class 2 and Class 3 SSCs and their associated supports; and the results of non PRA evaluations that are based on the IPEEE Screening Assessment for External Hazards, i.e., seismic margin analysis (SMA) to evaluate seismic risk, and a screening of other external hazards updated using the criteria in the endorsed ASME/ANS RA-Sa-2009

PRA Standard for other external hazard screening significance; as specified in Unit 1 [Unit 2] License Amendment [Number].

(2) Prior to implementation of the provisions of 10CFR 50.69, TVA shall complete the items below;

a. Items listed in Enclosure 1, Attachment 1, "SQN 10 CFR 50.69 PRA Implementation Items," in TVA letter CNL-19-002, "Response to Request for Additional Information Regarding Application to Modify Sequoyah Nuclear Plant Units 1 and 2, Application to Adopt 10 CFR 50.69, "Risk-informed Categorization and Treatment of Structures, Systems, and Components for Nuclear Power Reactors, (SQN-TS-17-06) (EPID: L-2018-LLA-0066)," dated March 21, 2019.

(3) Prior NRC approval, under 10 CFR 50.90, is required for a change to the categorization process specified above (e.g., change from a seismic margins approach to a seismic probabilistic risk assessment approach, change from alternative method for internal fire to a fire probabilistic risk assessment approach).

The NRC staff finds that the proposed license condition and referenced implementation items are acceptable because they adequately implement 10 CFR 50.69 using models, methods, and approaches consistent with the applicable guidance that has previously been endorsed by the NRC. The NRC staff, through an onsite audit or during future inspections, may choose to examine the closure of the implementation items with the expectation that any variations discovered during this review, or concerns with regard to adequate completion of the implementation item, will be tracked and dispositioned appropriately in accordance with the requirements of 10 CFR 50.69(f) and 10 CFR Part 50, Appendix B Criterion VI, and could be subject to NRC enforcement action(s).

4.0 STATE CONSULTATION

In accordance with the Commission's regulations, the Tennessee State official was notified of the proposed issuance of the amendments on July 10, 2019. The State official did not provide comments.

5.0 ENVIRONMENTAL CONSIDERATION

The amendments change a requirement with respect to installation or use of a facility component located within the restricted area as defined in 10 CFR Part 20. The NRC staff has determined that the amendments involve no significant increase in the amounts, and no significant change in the types, of any effluents that may be released offsite, and that there is no significant increase in individual or cumulative occupational radiation exposure. The Commission has previously issued a proposed finding that the amendments involve no significant hazards consideration, published in the *Federal Register* on August 28, 2018 (83 FR 26107), and there has been no public comment on such finding. Accordingly, the amendments meet the eligibility criteria for categorical exclusion set forth in 10 CFR 51.22(c)(9). Pursuant to 10 CFR 51.22(b), no environmental impact statement or environmental assessment need be prepared in connection with the issuance of the amendments.

6.0 CONCLUSION

The Commission has concluded, based on the considerations discussed above, that: (1) there is reasonable assurance that the health and safety of the public will not be endangered by operation in the proposed manner, (2) there is reasonable assurance that such activities will be conducted in compliance with the Commission's regulations, and (3) the issuance of the amendments will not be inimical to the common defense and security or to the health and safety of the public.

7.0 REFERENCES

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Date: September 18, 2019

SUBJECT: SEQUOYAH NUCLEAR PLANT, UNITS 1 AND 2 - ISSUANCE OF AMENDMENT NOS. 346 AND 340 RE: REQUEST TO ADOPT 10 CFR 50.69, "RISK-INFORMED CATEGORIZATION AND TREATMENT OF STRUCTURES, SYSTEMS, AND COMPONENTS FOR NUCLEAR POWER REACTORS" (EPID L-2018-LLA-0066) DATED SEPTEMBER 18, 2019

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