TVA TENNESSEE VALLEY AUTHORITY

1101 Market Street, Chattanooga, Tennessee 37402

CNL-21-085

February 24, 2022

10 CFR 50.90

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

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

Subject: **Sequoyah Nuclear Plant, Units 1 and 2, License Amendment Request to Modify Approved 10 CFR 50.69 Categorization Process (SQN-TS-21-07)**

Reference: Letter from NRC to TVA, "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 (ML19179A135)

In accordance with the provisions of Title 10 of the *Code of Federal Regulations* (10 CFR) 50.90, "Application for Amendment of License, Construction Permit, or Early Site Permit," Tennessee Valley Authority (TVA) is submitting a request for an amendment to the Operating License (OL) for the Sequoyah Nuclear Plant (SQN), Units 1 and 2.

The proposed amendment would modify the SQN OL to permit the use of the peer reviewed plant-specific SQN seismic probabilistic risk assessment (SPRA) and fire probabilistic risk assessment (FPRA) models into the previously approved 10 CFR 50.69 categorization process (Reference). In the referenced letter, License Conditions 33 and 27 (for SQN Units 1 and 2, respectively) specify that Nuclear Regulatory Commission (NRC) prior approval, under 10 CFR 50.90, is required for a change to a categorization process that is outside the bounds specified (e.g., a change from a seismic margins approach to an SPRA approach, or a change from an alternate method of internal fire approach to an FPRA approach). The scope of this request is limited to the change from the seismic margins approach to SPRA, and alternate method of internal fire approach to FPRA. No other changes to the categorization process are being requested by this license amendment request.

The Enclosure provides the basis for the proposed change to the SQN Unit 1 and 2 OL. Attachment 1 to the Enclosure identifies the current applicable SPRA and FPRA models. Attachment 2 to the Enclosure provides the markup to the SQN OLs. Attachment 3 to the Enclosure provides the final retyped pages to the SQN OLs.

TVA requests approval of the proposed license amendment within one year from the date of this submittal with implementation within 90-days following NRC approval.

TVA has determined that there are no significant hazards considerations associated with the proposed change and that the OL change qualifies for a categorical exclusion from environmental review pursuant to the provisions of 10 CFR 51.22(c)(9). In accordance with 10 CFR 50.91, “Notice for Public Comment; State Consultation,” a copy of this application with attachments is being provided to the Tennessee Department of Environment and Conservation.

There are no new regulatory commitments made in this letter. Please address any questions regarding this submittal to Stuart L. Rymer, Senior Manager, Fleet Licensing at slrymer@tva.gov.

I declare under penalty of perjury that the foregoing is true and correct. Executed on this 24th day of February 2022.

Respectfully,

Digitally signed by Carla Edmondson
Date: 2022.02.24 13:21:46 -05'00'

James T. Polickoski
Director, Nuclear Regulatory Affairs

Enclosure

Evaluation of the Proposed Change

cc (w/Enclosure):

NRC Regional Administrator – Region II
NRC Senior Resident Inspector – Sequoyah Nuclear Plant
NRC Project Manager – Sequoyah Nuclear Plant
Director, Division of Radiological Health – Tennessee State Department of Environment and Conservation

Enclosure

Evaluation of the Proposed Change
(25 pages)

Enclosure

Evaluation of the Proposed Change

Subject: **Sequoyah Nuclear Plant, Units 1 and 2, License Amendment Request to Modify Approved 10 CFR 50.69 Categorization Process (SQN-TS-21-07)**

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1. Current Applicable SPRA and FPRA Models
2. Proposed Operating License Changes (Units 1 and 2 Markups)
3. Proposed Operating License Pages (Units 1 and 2 Final Retyped)

Enclosure

1.0 SUMMARY DESCRIPTION

The proposed amendment would modify the licensing basis to implement a change to the approved voluntary implementation of the provisions of Title 10 of the *Code of Federal Regulations* (10 CFR), Part 50.69, "Risk-Informed Categorization and Treatment of Structures, Systems, and Components (SSCs) for Nuclear Power Plants." The proposed amendment would incorporate the use of the peer reviewed, plant-specific Sequoyah Nuclear Plant (SQN) Seismic Probabilistic Risk Assessment (SPRA) and Fire Probabilistic Risk Assessment (FPRA) models into the previously approved 10 CFR 50.69 categorization process with License Amendments 346 and 340 (Reference 1), as allowed by the Nuclear Regulatory Commission (NRC) endorsed industry guidance. The provisions of 10 CFR 50.69 allow adjustment of the scope of equipment subject to special treatment controls (e.g., quality assurance, testing, inspection, condition monitoring, assessment, and evaluation). For equipment determined to be of low safety significance, alternative treatment requirements can be implemented in accordance with this regulation. For equipment determined to be of high safety significance, requirements will either not be changed, or will be enhanced. This allows improved focus on equipment that has safety significance resulting in improved plant safety.

The 10 CFR 50.69 categorization process has been approved by the NRC for SQN in Reference 1. Categorization includes an integrated assessment of total risk and the regulations and categorization guidance allows licensees to implement different approaches depending on the scope of their probabilistic risk assessment (PRA) models. The currently approved risk assessment tools are listed below.

1. Internal Events/Internal Flooding Probabilistic Risk Assessment (PRA) models for internal risk
2. Seismic Safe Shutdown Equipment List (SSSEL) from the seismic margin analysis (SMA) for seismic risk
3. Fire Safe Shutdown Equipment List (FSSEL) from the Appendix R analysis for fire risk as documented in the SQN Fire Protection Report referenced in the Updated Final Safety Analysis Report
4. Hazard Screening in accordance with American Society of Mechanical Engineers (ASME)/American Nuclear Society (ANS) RA-Sa-2009 Part 6 to assess risk from other external hazards (high winds, external floods, etc.)
5. Assess shutdown risk using NUMARC 91-06

This proposed amendment request substitutes a peer reviewed SPRA model in place of the SSSEL to assess seismic risk and substitutes a peer reviewed FPRA in place of using the FSSEL to assess fire risk. This type of change was envisioned by the regulations and guidance as new Probabilistic Risk Assessment (PRA) tools became available. All other aspects of the program remain as the NRC approved in Reference 1.[1]

[1] Note – No categorizations have been implemented to date under the existing approved methods, pending completion of the Early Warning Time re-analysis for the external flooding event, as described in Section 3.5.3.3 of Reference 1.

2.0 DETAILED DESCRIPTION

2.1 Current Regulatory Requirements

The NRC has established a set of regulatory requirements for commercial nuclear reactors to ensure that a reactor facility does not impose an undue risk to the health and safety of the public, thereby providing reasonable assurance of adequate protection to public health and safety. The current body of NRC regulations and their implementation are largely based on a "deterministic" approach.

This deterministic approach establishes requirements for engineering margin and quality assurance in design, manufacture, and construction. In addition, it assumes that adverse conditions can exist (e.g., equipment failures and human errors) and establishes a specific set of design basis events (DBE). The deterministic approach then requires that the facility include safety systems capable of preventing or mitigating the consequences of those DBEs to protect public health and safety. Those SSCs necessary to defend against the DBEs are defined as "safety-related," and these SSCs are the subject of many regulatory requirements, herein referred to as "special treatments," designed to ensure that they are of high quality, high reliability, and have the capability to perform during postulated design basis conditions.

Special treatment includes, but is not limited to, quality assurance, testing, inspection, condition monitoring, assessment, evaluation, and resolution of deviations. The distinction between "treatment" and "special treatment" is the degree of NRC specification as to what must be implemented for particular SSCs or for particular conditions. Typically, the regulations establish the scope of SSCs that receive special treatment using one of three different terms: "safety-related," "important to safety," or "basic component." The terms "safety-related" and "basic component" are defined in the regulations, while "important to safety," used principally in the general design criteria (GDC) of Appendix A to 10 CFR Part 50, is not explicitly defined.

The previously approved SQN 50.69 categorization process conforms to the guidance in NRC Regulatory Guide (RG) 1.201, "Guidelines for Categorizing Structures, Systems, and Components in Nuclear Power Plants According to their Safety Significance," Revision 1, dated May 2006 (Reference 2). The categorization process also conforms to the guidance in Nuclear Energy Institute (NEI) 00-04, "10 CFR 50.69 SSC Categorization Guideline," Revision 0, dated July 2005 (Reference 3), as endorsed by RG 1.201. With this change, to utilize the SPRA and FPRA models rather than assuming that all components identified in the SSSEL and FSSEL have high safety significance (HSS), the SQN categorization process will continue to conform to these guidance documents.

2.2 Reason for Proposed Change

A probabilistic approach to regulation enhances and extends the traditional deterministic approach by allowing consideration of a broader set of potential challenges to safety, provides a logical means for prioritizing these challenges based on safety significance, and allows consideration of a broader set of resources to defend against these challenges. In contrast to the deterministic approach, PRA addresses credible initiating events by assessing the event frequency. Mitigating system reliability is then assessed, including the potential for common cause failures. The probabilistic approach to regulation is an extension and enhancement of traditional regulation by considering risk in a comprehensive manner. To take advantage of the safety enhancements available through the use of PRA, in 2004 the NRC published a new regulation, 10 CFR 50.69. The provisions of 10 CFR 50.69 allow adjustment of the scope of equipment subject to special treatment controls (e.g., quality assurance, testing, inspection, condition monitoring, assessment, and evaluation). For equipment determined to be of low safety significance (LSS), alternative treatment requirements can be implemented in accordance with the regulation. For equipment determined to be HSS, requirements will not be changed or will be enhanced. This allows improved focus on HSS equipment resulting in improved plant safety.

The rule contains requirements on 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 classification (RISC) categories. The determination of safety significance is performed by an integrated decision-making process, as described in Reference 3, 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 how SSC is categorized. Finally, assessment activities are conducted to make adjustments to the categorization and treatment processes as needed so that SSCs continue to meet all applicable requirements.

The rule 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, the rule enables licensees 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, the rule allows for alternative treatments that provide reasonable confidence that these SSCs will satisfy functional requirements. Implementation of 10 CFR 50.69 allows an improved focus on equipment that has safety significance resulting in improved plant safety.

The SQN 10 CFR 50.69 categorization process has been previously approved by the NRC in Reference 1. The proposed change implements a modification to the process, as allowed by the 10 CFR 50.69 guidance endorsed by NRC in RG 1.201 (Reference 3), to incorporate use of the peer reviewed plant-specific SQN SPRA and FPRA models.

2.3 Description of the Proposed Change

Tennessee Valley Authority (TVA) proposes the revision of License Condition C.(33) (SQN Unit 1) and C.(27) (SQN Unit 2) as provided in Attachment 1. Prior NRC approval, under 10 CFR 50.90, is required for a change to the categorization process, as specified in Paragraph (3) of each License Condition.

Note – The change to Paragraph (3) within the above License Conditions to cite a different example of a change to the categorization process to replace reference changes to an SPRA and FPRA approach requiring NRC prior approval under 10 CFR 50.90, is considered to be an administrative change.

3.0 TECHNICAL EVALUATION

As 10 CFR 50.69 specifies the information to be provided by a licensee requesting adoption of the regulation, this request conforms to the requirements of 10 CFR 50.69(b)(2), which states:

> A licensee voluntarily choosing to implement this section shall submit an application for license amendment under 10 CFR 50.90 that contains the following information.
>
> i. A description of the process for categorization of RISC-1, RISC-2, RISC-3, and RISC-4 SSCs.
> ii. A description of the measures taken to assure that the quality and level of detail of the systematic processes that evaluate the plant for internal and external events during normal operation, low power, and shutdown (including the plant-specific probabilistic risk assessment (PRA), margins-type approaches, or other systematic evaluation techniques used to evaluate severe accident vulnerabilities) are adequate for the categorization of SSCs.
> iii. Results of the PRA review process conducted to meet 10 CFR 50.69(c)(1)(i).
> iv. A description of, and basis for acceptability of, the evaluations to be conducted to satisfy 10 CFR 50.69(c)(1)(iv). The evaluations must include the effects of common cause interaction susceptibility, and the potential impacts from known degradation mechanisms for both active and passive functions, and address internally and externally initiated events and plant operating modes (e.g., full power and shutdown conditions).

The above information was previously provided to NRC as part of the SQN license amendment request (Reference 4). The SQN 10 CFR 50.69 categorization process (overall process, including active and passive categorization elements) was approved by the NRC in Reference 1. In its review and approval of that application, NRC reviewed the technical acceptability of the SQN internal events at power PRA model and approved their use for 10 CFR 50.69 categorization. The SQN 10 CFR 50.69 process currently addresses seismic and fire risk through the use of the SSSEL and FSSEL. The purpose of this license amendment request is to replace, within the approved SQN 10 CFR 50.69 program, the use of the SSSEL process with use of the SQN SPRA and the use of the FSSEL process with use of the SQN FPRA in accordance with NEI 00-04 (Reference 3) and RG 1.201 (Reference 2) guidance. Therefore, the remainder of this technical evaluation is focused on establishing the technical adequacy of the SQN SPRA and FPRA for this application.

3.1 <u>SPRA Technical Adequacy Evaluation (10 CFR 50.69(B)(2)(11))</u>

The following sections demonstrate that the quality and level of detail of the processes used in categorization of SSCs with respect to seismic hazards using the SPRA are acceptable. The SPRA model described below has been peer reviewed and there are no PRA upgrades that have not been peer reviewed. There are no open Finding level Facts and Observations (F&O).

3.1.1 <u>Seismic Hazards</u>

The approved SQN categorization process uses the SSSEL from the seismic margin analysis (SMA) performed for the individual plant examination for external events (IPEEE) in response to Generic Letter (GL) 88-20 (Reference 5) for evaluation of safety significance related to seismic hazards. If a component is credited on the SSSEL, it is considered HSS. Through this requested change, the SQN categorization process will instead use the peer reviewed plant-specific SQN SPRA model. The TVA risk management process ensures that the SPRA model used in this application reflects the as-built and as-operated plant for each of the SQN units. No plant specific approaches were utilized in development of the seismic hazards for the SPRA model. Attachment 1 to this enclosure identifies the current applicable SPRA model.

Expert judgment, as defined in Section 1-4.3 of American Society of Mechanical Engineers / American Nuclear Society (ASME/ANS) RA-Sb-2013, was used in the probabilistic seismic hazard analysis (PSHA). As required by the PRA Standard, the frequency of occurrence of earthquake ground motions at the site was based on the PSHA. The seismic source characterization inputs to the PSHA are based on the Central and Eastern U.S. (CEUS) regional seismic source characterization model published in NUREG-2115 (Reference 6) (i.e., the "CEUS-SSC" model), with updates described in Electric Power Research Institute (EPRI), 2015 (Reference 7). The ground motion characterization (GMC) inputs to the PSHA are based on an updated model published in 2013 by EPRI's CEUS ground motion update project (EPRI, 2013a and 2013b, References 8 and 9). The seismic hazard analysis for the Sequoyah site also accounts for the effects of local site response. The Senior Seismic Hazard Analysis Committee (SSHAC) methodology defines a process of structured expert interaction (elicitation) that is considered a minimum technical requirement for conduct of a PSHA.

The SSHAC process of NUREG/CR-6372 (Reference 10), and the NUREG-2117 (Reference 11) process of conducting a PSHA were used to develop both the seismic source characterization and GMC models used as inputs to the analysis. Use of the SSHAC methodology ensures that data, methods, and models supporting the PSHA are fully incorporated and that uncertainties are fully considered in the process at sufficient depth and detail necessary to satisfy scientific and regulatory needs. The SSHAC-related guidance documents define and describe four "levels." The level of study is not mandated in the Standard; however, both the seismic source characterization and the GMC parts of the PSHA were developed using a SSHAC Level 3 analyses. In the case of the GMC, a SSHAC Level 2 analysis was carried out to update a prior Level 3 study. These Level 3 studies satisfy the requirements of the Standard related to the method of conduct of the PSHA generally, as well as addressing several individual requirements related to data collection, data evaluation and model development, and quantification of uncertainties.

3.1.2 SPRA Peer Review

The SQN Units 1 and 2 SPRA was peer reviewed against the requirements of Part 5 of ASME/ANS RA-Sb-2013 (Addendum B), the ASME/ANS PRA Standard (Reference 12). The peer reviewer qualifications have been reviewed by TVA and have been confirmed to be consistent with requirements in the ASME/ANS PRA Standard and the guidelines of NEI 12-13. The SPRA peer reviewers had no previous involvement in the SQN Units 1 & 2 SPRA. This is certified by the reviewers' signatures on the cover of the peer review report. This satisfies the independence requirements of Section 1-6.2.2 of the ASME/ANS PRA Standard.

This peer review was performed using the process defined in NEI guidelines NEI 12-13 (Reference 13) as amended by the Nuclear Regulatory Commission (NRC) in March of 2018 (Reference 14).

Section 5 of the ASME/ANS PRA Standard (Reference 12) contains a total of 77 supporting requirements (SR) under 3 technical elements. The SQN SPRA was reviewed against Capability Category (CC) II of the PRA Standard for all applicable SRs. Five (5) of the SRs were judged to be not applicable for SQN, and therefore the remaining 72 SRs were reviewed.

The SQN Units 1 & 2 Internal Events PRA was peer reviewed in 2011 and all the findings from that review were closed in 2017 following the process in Appendix X of the NEI peer review guidance documents recently accepted by the NRC (Reference 15).

Of the 72 SRs reviewed, two were NOT MET and 70 were MET at CC-II or greater. As a result of this review, a total of 53 unique Finding level F&Os were generated. There were no "unreviewed analysis methods" identified during the review.

3.1.3 SPRA Peer Review F&O Closure Review

The SQN SPRA F&O Independent Assessment & Focused-Scope Peer Review was performed at the TVA corporate offices in Chattanooga, Tennessee, from February 4 through 8, 2019. The purpose was to perform an independent assessment in accordance with Appendix X of NEI 05-04, 07-12, and 12-13 (Reference 16) to review TVA's proposed close out of 53 unique Finding level F&Os of record from the prior PRA peer review (discussed above) against the ASME/ANS RA-Sb-2013 PRA Standard and to perform a Focused-Scope peer review for an Upgrade to the SPRA. The Appendix X process has been accepted by the NRC for use (Reference 17).

The Independent Assessment Team consisted of six team members with extensive qualifications and many years of experience in all areas of SPRA. All reviewers met the criteria specified in NEI 05-04 and NEI 12-13 and the ASME/ANS RA-Sa-2009 PRA Standard Section 1-6.2, and in NRC's Reference 17 memoranda outlining expectations for a finding closure Independent Assessment.

Four of the seven seismic hazards analysis (SHA) findings were assessed to be upgrades and were resolved as part of a focused scope review of High Level Requirements SHA-I and SHA-J. Following completion of the Independent Assessment and focused scope review, all of the original 53 Finding level F&Os were closed and the associated SRs all meet CC II or greater. There were no "unreviewed analysis methods" identified during the focused scope peer review. With the closure of all peer review findings, the SPRA model of record meets the requirements for PRA technical acceptability for this application. No upgrades were required as a part of the F&O closure review.

3.1.4 Discussion of the Use of Addendum B of the PRA Standard

In 2013, the ASME/ANS PRA Standard was revised by the ASME/ANS Joint Committee on Nuclear Risk Management (JCNRM). This standard, ASME/ANS RA-Sb-2013 (Addendum B), was approved by ANSI in 2013, but has not been formally endorsed by the NRC through a revision to RG 1.200, "An Approach for Determining the Technical Acceptability of Probabilistic Risk Assessment Results for Risk-Informed Activities." TVA has performed a gap analysis from the NRC-endorsed 2009 standard consistent with the information provided in a Southern Nuclear Operating Company, Inc., letter to NRC, for the Vogtle Electric Generating Plant, Units 1 and 2 (Reference 18). NRC acceptance of the assessment was documented in a letter to Southern Nuclear Operating Company (Reference 19).

In the Vogtle Assessment, all but six of the Addendum B SRs have been shown to be equal to the corresponding Addendum A SRs, or have been shown to envelop the corresponding Addendum A SRs. The remaining six SRs (SHA-B3, SHA-C3, SFR-C3, SFR-C6, SFR-G3, and SPR-B1) show that the SQN SPRA conforms to these Addendum A SRs. This is discussed in the Table E-1 below. Based on References 18 and 19 and the table shown below, the peer review of the SPRA using Addendum B is acceptable.

Table E-1

<table>
<tr><th colspan="5">Supplemental Information on Supporting Requirements for SQN</th></tr>
<tr><th>SR</th><th>CC-I</th><th>CC-II</th><th>CC-III</th><th>Basis for Assessment</th></tr>
<tr><td rowspan="2">SHA-B3</td><td>Addendum A

CC-I is not a focus of the assessment.</td><td colspan="2">Addendum A

As part of data collection, COMPILE a catalog of historically reported, geologically identified, and instrumentally recorded earthquakes. USE reference requirements or equivalent.</td><td rowspan="2">SQN Conforms to Addendum A
The difference between the Addendum A and B requirements is that Addendum A required a catalog of seismic data be compiled opposed to Addendum B that required inclusion of an existing catalog of seismic data.

Subsequent to Addendum A an extensive seismic database of information was available from a study referred to as the CEUS seismic source characterization study. This study considered the full range of earthquake data (geologically identified, seismologic (instrumentally recorded), geophysical and tectonic) to develop the CEUS seismic source characterization model (EPRI/DOE/NRC, 2012). The data collection effort included the compilation of an earthquake catalog.

The level of study is not mandated in the Standard. However, both the seismic source characterization and the Ground Motion Model (GMM) parts of the Probabilistic Seismic Hazard Assessment (PSHA) were developed as a result of SSHAC Level 3 analyses. In the case of the GMM, a SSHAC Level 2 analysis was carried out to update a prior Level 3 study.

These Level 3 studies satisfy the requirements of the Standard. As a first step to performing a PSHA, the Standard requires that an up-to-date database, including regional geological, seismological, geophysical data, and local site topography, and a compilation of information on surficial geologic and geotechnical site properties. These data include a catalog of relevant</td></tr>
<tr><td colspan="2">Addendum B

INCLUDE an appropriate existing catalog of historically reported earthquakes, instrumentally recorded earthquakes, and earthquakes reported through geological investigations. USE reference requirements or equivalent.</td><td>Addendum B

CC-III is not a focus of the assessment.</td></tr>
</table>

Supplemental Information on Supporting Requirements for SQN				
SR	**CC-I**	**CC-II**	**CC-III**	**Basis for Assessment**
				historical, instrumental, and paleoseismic information within 320 km of the site. The CEUS seismic source characterization study involved an extensive data collection effort that satisfies the requirements of the Standard as it relates to developing a regional-scale seismic source model. The 2012 CEUS seismic source characterization catalog followed a SSHAC Level 3 process, and is applicable for risk informed applications. Compiling a new catalog will not be as rigorous as the SSHAC Level 3 process. The Addenda B SR is appropriate for CC-II. The 2012 CEUS seismic source characterization report used an earthquake catalog, which extended through 2008. Recent earthquake activity in the vicinity of the SQN site was assessed for its impact on hazard. Based on this, the SQN PSHA that was performed conforms to Addendum A.
SHA-C3	Addendum A CC-I is not a focus of the assessment.	Addendum A The seismic sources are characterized by source location and geometry, maximum earthquake magnitude, and earthquake recurrence. INCLUDE the aleatory and epistemic uncertainties explicitly in these characterizations.		**SQN Conforms to Addendum A** Addenda B added additional clarification into the text of this SR, and also added a clause "where significant" at the end. The Addenda B SR is appropriate for CC-II. Under the SSHAC Level 3 process, the aleatory and epistemic uncertainties in seismic sources are characterized for source location and geometry, magnitude, and activity rate. Logic trees to account for the epistemic uncertainty were developed as part of the SSHAC Level 3 methodology implemented in the CEUS seismic source characterization report. Aleatory variability and epistemic uncertainty are handled separately in a manner consistent with the seismic source characterization and GMC components of the PSHA. Aleatory variability is modeled via randomizations of the shear wave
	Addendum B CC-I is not a focus of the assessment.	Addendum B The seismic sources are characterized by alternative source representation and source geometry, maximum earthquake magnitude, and earthquake recurrence. INCLUDE the aleatory and epistemic uncertainties explicitly in these characterizations, where significant.		

Supplemental Information on Supporting Requirements for SQN				
SR	**CC-I**	**CC-II**	**CC-III**	**Basis for Assessment**
				velocity profile and nonlinear properties (i.e., modulus reduction and damping curves) that are used to develop a median and standard plus deviation of the amplification factor for a particular set of base case properties. Epistemic uncertainty in shear wave velocity profile, kappa, nonlinear properties, single- vs. double corner source model, and magnitude are incorporated via a logic tree with appropriate weights on each branch. Based on the discussion above, the SQN PSHA that was performed conforms to Addendum A.
SFR-C3	Addendum A If scaling of existing design response analysis is used, JUSTIFY it based on the adequacy of structural models, foundation characteristics, and similarity of input ground motion.		Addendum A CC-III is not a focus of the assessment.	**SQN Conforms to Addendum A** The change from Addendum A to Addendum B involved the deletion of the word "design" from “existing design response analysis.” Scaling of the fragility evaluation of the RCS loop piping was performed since direct responses could not be obtained from the (State-of-the-Art Soil-Structure Interaction) SASSI response analysis. An enhanced scaling approach based upon SASSI transfer functions was applied to arrive at a more refined design basis scaling high confidence of low probability of failure (HCLPF) that properly incorporates multi-mode effects.
	Addendum B If scaling of existing response analysis is used, JUSTIFY it based on the adequacy of structural models, foundation characteristics, and similarity of input ground motion.		Addendum B CC-III is not a focus of the assessment.	
SFR-C6	Addendum A When soil-structure interaction (SSI) analysis is conducted, ENSURE that it is median centered using median properties, at soil strain levels corresponding to the input ground motions that dominate the seismically induced core damage frequency. CCOUNT for the uncertainties in the SSI analysis by varying the low strain soil shear modulus between the median value times *(1 + Cv)* and the median value divided by *(1 + Cv)*, where Cv is a factor that accounts for uncertainties in		Addendum B CC-III is not a focus of the assessment.	**SQN Conforms to Addendum A** The changes in SFR-C6 involved the replacement of "ACCOUNT for" with the more precise action verb "INCLUDE", the nonsubstantive replacement of "dominate" with "contribute most" for PRA standard consistency, and the removal of how to perform SSI uncertainty analysis. The SSI uncertainty analysis method presented in Addendum A is derived from American Society of Civil Engineers (ASCE) 4-98 (as indicated by the nonmandatory Note 5).

Supplemental Information on Supporting Requirements for SQN				
SR	**CC-I**	**CC-II**	**CC-III**	**Basis for Assessment**
	the SSI analysis and soil properties. If adequate soil investigation data are available, ESTABLISH the mean and standard deviation of the low strain shear modulus for every soil layer. Then ESTABLISH the value of Cv so that it will cover the mean plus or minus one standard deviation for every layer. The minimum value of Cv is 0-5. When insufficient data are available to address uncertainties in soil properties, ENSURE that Cv is taken as no less than 1.0.			Section 3.3.1.7 of ASCE 4-98 states that the use of *(1 + Cv)* to vary low strain soil shear moduli is an acceptable method *in lieu of probabilistic evaluation,* which Section C.3.3.1.7 further states is the preferred approach. The site response analysis, which provides the basis for the inputs to the SSI analysis, accounts for variabilities of the ground motion input at hard rock and uncertainties in the Vs profile. Uncertainties in the hard rock motion are accounted for by considering a suite of hard rock time histories. Uncertainties in the Vs profile are propagated by considering random profiles obtained by varying the low strain best estimate profile using layer Cv based on data from site investigations. The effects of the uncertainties in the resulting strain compatible Vs profile on the SSI response are accounted for by considering *(1+Cv)* and *(1Cv)* times the median s train compatible Vs profile, with a minimum *Cv = 0.5*. The Cv are based on the site response analysis, performed as part of the PSHA/GMRS (ground motion response spectrum) development. Only the best estimate response is evaluated. This analysis is median based and uses the median subsurface shear wave velocity profile (consistent with soil/rock strain levels associated with the input spectra), and best estimate structure properties such as stiffness and damping associated with the expected level of damage.
	Addendum B When soil-structure interaction (SSI) analysis is conducted, ENSURE that it is median centered using median properties, at soil strain levels corresponding to the input ground motions that contribute most to the seismically induced core damage frequency. INCLUDE the uncertainties in the SSI analysis.		Addendum B CC-III is not a focus of the assessment.	
SFR-G3	Addendum A DOCUMENT the sources of model uncertainty and related assumptions associated with the seismic fragility analysis.			**SQN Conforms to Addendum A** Addendum B deleted this SR. However, the SQN SPRA documentation describes in detail the sources of model uncertainty and related assumptions associated with the seismic fragility analysis. Therefore, the SQN SPRA conforms to Addendum A.
	Addendum B Deleted			
SPR-B1	Addendum A			SQN conforms to Addendum A. Addendum B removed the last

Supplemental Information on Supporting Requirements for SQN				
SR	CC-I	CC-II	CC-III	Basis for Assessment
	In each of the following aspects of the seismic-PRA systems-analysis work, SATISFY the corresponding requirements in Part 2, except where they are not applicable or where this Part includes additional requirements. DEVELOP a defined basis to support the claimed non-applicability of any exceptions. The aspects governed by the requirements are: (a) initiating-event analysis (b) accident-sequence analysis (c) success-criteria analysis (d) systems analysis (e) data analysis (f) human-reliability analysis (g) use of expert judgement When the Part 2 requirements are used, FOLLOW the Capability Category designations in Part 2, and for consistency USE the same Capability Category in this analysis.			sentence of this SR in response to an EPRI 2011 comment on the Addendum B ballot. The last sentence was removed in Addendum B because it was determined to be confusing as well as inappropriate specificity to require all new aspects in the SPRA to meet the exact same CCs of Part 2 SRs. In addition, Addendum B changed the action verb to be consistent with accepted verb usage across SRs. The Addendum B SR clarifications are appropriate. Regardless, SQN SPRA builds upon the internal events PRA and uses the same general methodologies as used for Part 2 where applicable; therefore, the SQN SPRA conforms to Addendum A.
	Addendum B In each of the following aspects of the seismic-PRA systems-analysis work, SATISFY the corresponding requirements in Part 2, except where they are not applicable or where this Part includes additional requirements. SPECIFY a basis to support the claimed non-applicability of any exceptions. The aspects governed by this requirement are: (a) initiating-event analysis (b) accident-sequence analysis (c) success-criteria analysis (d) systems analysis (e) data analysis (f) human-reliability analysis (g) use of expert judgment			

3.2 FPRA Technical Adequacy Evaluation (10 CFR 50.69(b)(2)(11))

The following sections demonstrate that the quality and level of detail of the processes used in categorization of SSCs with respect to fire hazards using the FPRA are acceptable. The FPRA model described below has been peer reviewed and there are no PRA upgrades that have not been peer reviewed.

3.2.1 Fire Hazards

TVA is currently 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 the fire safe shutdown equipment list (Fire SSEL) in the SQN Fire Protection Report referenced in the Updated Final Safety Analysis Report to evaluate internal fire events. This approach ensures the SSCs that are credited to establish and maintain safe shutdown capability are retained as safety significant. If a component is credited on the Fire SSEL, it is considered HSS.

Through this requested change, the SQN categorization process will instead use the peer reviewed plant-specific SQN FPRA model. The TVA risk management process ensures that the FPRA model used in this application reflects the as-built and as-operated plant for each of the SQN units. No plant specific approaches were utilized in development of the fire hazards for the FPRA model. Attachment 1, at the end of this enclosure, identifies the current applicable FPRA model.

3.2.2 FPRA Peer Review

The SQN Units 1 & 2 Fire Probabilistic Risk Assessment (FPRA) was peer reviewed against the requirements of Part 4 of the ASME/ANS 2009 PRA Standard (Addendum A, Reference 20). The peer review also included the clarifications and qualifications provided in the NRC endorsement of the PRA Standard, contained in Revision 2 to RG 1.200 (Reference 21).

The peer reviewer qualifications have been reviewed by TVA and have been confirmed to be consistent with requirements in the ASME/ANS PRA Standard and the guidelines of NEI 07-12. This peer review was performed using the process defined in NEI 07-12 (Reference 22). The FPRA peer reviewers had no previous involvement in the SQN Units 1 & 2 FPRA. This is certified by the reviewers' signatures on the cover of the peer review report. This satisfies the independence requirements of Section 1-6.2.2 of the ASME/ANS PRA Standard.

Part 4 (Fire) of the ASME/ANS PRA Standard contains a total of 173 SRs under 13 technical elements. The SQN FPRA was reviewed against CC II of the PRA Standard for all applicable SRs. A total of 25 of the supporting requirements were judged by the peer team to be not applicable to SQN, and therefore, the remaining 148 SRs were reviewed.

The peer review team concluded, in general, the data, methodologies and fire risk models used for SQN Units 1 & 2 were appropriate and sufficient to meet the Standard. As noted the peer review report, the SRs were met at Category II or higher for all but nine of the 148 supporting requirements applicable to the SQN FPRA. Six SRs were NOT MET and three were MET at CC-I. In the judgment of the peer review team, the SQN FPRA meets the remaining supporting requirements based on the current FPRA methodology utilized, the FPRA models and results, and the detailed documentation. All methods used in the SQN FPRA align with NRC-endorsed methodologies. The review team identified specific areas for improving the quality of the FPRA. These areas are documented as F&Os including 32 findings. There were no “unreviewed analysis methods” identified during the review.

3.2.3 FPRA Peer Review F&O Closure Review

The SQN FPRA F&O Closure by Independent Assessment (F&O closure assessment) was held during the week of April 27, 2020. The purpose of the meeting was to perform an independent F&O closure assessment in accordance with NEI 07-12 Appendix X to review close out of Finding level F&Os of record from the prior PRA peer review.

The F&O closure assessment was based on the results of the most recently completed FPRA peer review discussed above. F&O dispositions reviewed and determined to have been adequately addressed through this F&O closure assessment are considered “closed” and no longer relevant to the current PRA model.

The SQN FPRA F&O closure independent assessment team was comprised of five team members with extensive qualifications and many years of experience in relevant areas of fire probabilistic risk assessment (FPRA). All reviewers met the criteria specified in NEI 07-12, and the ASME/ANS PRA Standard (Reference 20) Section 1-6.2. The 32 Finding level F&Os were reviewed and are assessed as closed. No F&Os were assessed as partially closed or open and the SRs that the previous peer review had found to be NOT MET or MET at CC-I, are now assessed as MET at CC-II or better. With the closure of all peer review findings, the FPRA model of record meets the requirements for PRA technical acceptability for this application. No upgrades were required as a part of the F&O closure review.

3.3 PRA Maintenance and Updates

The TVA risk management process, which was previously reviewed by NRC as part of the SQN 10 CFR 50.69 approval (Reference 1), ensures that the applicable PRA models, including the SPRA and FPRA models, used in this application continue to reflect the as-built and as-operated plant for each of the SQN units. The process delineates the responsibilities and guidelines for updating the PRA models and includes criteria for both regularly scheduled and interim PRA model updates. The process includes provisions for monitoring potential areas affecting the PRA models (e.g., due to changes in the plant, errors or limitations identified in the model, industry operational experience) for assessing the risk impact of unincorporated changes, and for controlling the model and associated computer files. The process will assess the impact of these changes on the plant PRA model in a timely manner but no longer than once every two refueling outages. If there is a significant impact on the PRA model, the SSC categorization will be re-evaluated.

In addition, TVA has implemented a process that addresses the requirements in NEI 00-04 (Reference 3), Section 11, "Program Documentation and Change Control." The process reviews the results of periodic and interim updates of the plant PRA that may affect the results of the categorization process. If the results are affected, adjustments will be made as necessary to the categorization or treatment processes to maintain the validity of the processes. In addition, any PRA model upgrades to any of the PRA models used in support of the SQN 10 CFR 50.69 process will be peer reviewed prior to implementing those changes in the PRA model used for categorization.

3.4 Modeling of FLEX Equipment in the SQN PRA Models

The SQN PRA Models (internal Events and Internal Flooding, Fire, and Seismic) include the two 480V FLEX diesel generators (DGs) permanently installed on the auxiliary building roof, the two 6.9KV 3 MW FLEX DGs installed in the additional diesel generator building (ADGB) and the associated ventilation and fuel supply systems. There are no automatic starts modeled for any of the FLEX DGs. No portable FLEX equipment is credited in the models.

The 480V FLEX DGs are installed with an integral, protected day tank to provide a maximum of 10 hours of operation before fuel makeup is required. Refueling of the 480V DG day tank is achieved using the fuel oil transfer pump taking suction from the flood tank. The fuel for the 6.9KV 3 MW DGs is initially supplying from the day tank. The extended fuel supply is available from the 7-day tank via the fuel oil transfer pump.

The 480V FLEX DGs are secured inside rooms located on the auxiliary building roof where the maximum calculated temperature with the DGs on standby is within the design temperature. Therefore, no ventilation is modeled for the 480V FLEX DGs.

The ADGB room ventilation system is capable of adequate cooling of the 6.9KV 3 MW FLEX DG radiators and other skid mounted equipment. The sliding door (tornado door, SQN-0-DOOR-410-D042) is required to be open when a 6.9KV 3 MW FLEX DG is running to allow sufficient supply of aspirating and cooling air. Airflow through the ADGB to support 6.9KV 3 MW FLEX DG operation will be provided by the DG radiator fans and exhaust fans.

Operator actions to start and maintain (including refueling) the DGs are included in the models.

3.5 Use of Floor Values in the Human Reliability Dependency Analysis

The dependency analysis for the Internal Events/Internal Flooding PRA model was performed on the SQN Unit 1 and Unit 2 core damage frequency (CDF) and large early release frequency (LERF) cutset files. (Single unit files were combined into one file - Total and a single dependency analysis for both units completed.) This file also included a Human Reliability Analysis (HRA) from the Internal Flooding Analysis. This information was incorporated in the SQN quantification recovery rule file as described in the SQN PRA Quantification Notebook. After final quantification, application of the recovery rules accounts for the level of dependency determined by the HRA Calculator.

The recovery rule file was developed to limit the joint probability of each combination to be no less than 1.0E-05. This lower bound was selected from NUREG-1792 (Reference 23) because some of the Performance-Shaping Factors are global in nature and apply as a sum instead of a product. In order to satisfy this, the recovery rule file was developed to limit the joint probability of each combination to be no less than 1.0E-05.

The dependency analysis for the SQN Unit 1 and Unit 2 Fire and SPRA models was performed similarly to what was done for the Internal Events/Internal Flooding model. The recovery rule file for each model was developed to limit the joint probability of each combination to be no less than 1.0E-05 in accordance with NUREG-1792.

3.6 PRA Uncertainty Analysis for the SPRA and FPRA Models

Section 3.3.2 of NRC RG 1.200 requires any application that calls on RG 1.200 to identify the key assumptions and approximations relevant to the application. RG 1.200 defines key source of uncertainty and key assumption as follows:

> A key source of uncertainty is one that is related to an issue in which there is no consensus approach or model and where the choice of approach or model is known to have an impact

on the risk profile (e.g., total core damage frequency (CDF) and total large early release frequency (LERF), the set of initiating events and accident sequences that contribute most to CDF and to LERF) such that it influences a decision being made using the PRA. Such an impact might occur, for example, by introducing a new functional accident sequence or a change to the overall CDF or LERF estimates significant enough to affect insights gained from the PRA.

A key assumption is one that is made in response to a key source of model uncertainty in the knowledge that a different reasonable alternative assumption would produce different results, or an assumption that results in an approximation made for modeling convenience in the knowledge that a more detailed model would produce different results. For the base PRA, the term "different results" refers to a change in the risk profile (e.g., total CDF and total LERF, the set of initiating events and accident sequences that contribute most to CDF and to LERF) and the associated changes in insights derived from the changes in the risk profile. A "reasonable alternative" assumption is one that has broad acceptance within the technical community and for which the technical basis for consideration is at least as sound as that of the assumption being challenged.

This section describes how assumptions and sources of uncertainty for the full power internal events PRA (includes internal floods), FPRA, and SPRA were assessed to identify any key sources of uncertainty or key assumptions. A comprehensive list of assumptions and sources of uncertainty is compiled, including those associated with plant-specific features, modeling choices, and generic industry concerns. A disposition is provided for each assumption and source of uncertainty, addressing the impact on the risk-informed application. The risk metrics of interest for the RICT application are CDF and LERF risk due to internal events including internal flooding, fire, and seismic events. For any potential key source of uncertainty or potential key assumption judged not to be key to the application, a discussion is provided to indicate why it does not need to be addressed further in the context of the application.

This evaluation involves two steps:

1. A review of the assumptions documented in the PRAs. The significance of each assumption is evaluated to determine whether it is a key source of uncertainty for the RICT application.

2. A review of the generic sources of uncertainty documented in EPRI Report 1016737 and EPRI Report 1026511. The significance of each generic source of uncertainty is evaluated to determine whether it is a key source of uncertainty for the RICT application.

To identify the assumptions and uncertainties used in the Internal Events PRA (includes internal floods), FPRA, and SPRA models supporting this application, the generic issues identified in Table A.1 of EPRI 1016737 and EPRI 1026511 were reviewed, as well as the PRA documentation for plant-specific assumptions and uncertainties. This identification process is consistent with NUREG-1855 Revision 1, Stage E: Assessing Model Uncertainty (Reference 24).

EPRI Report 1016737 focuses on uncertainties from internal events PRAs. The list of generic sources of uncertainty pertaining to internal events PRAs is given in Tables A-1 and A-2 of the EPRI report.

EPRI Report 1026511 focuses on uncertainties from (a) FPRAs, (b) SPRAs, (c) low power and shutdown (LPSD) PRAs, and (d) Level 2 PRAs. SQN does not have a LPSD PRA. This leaves generic sources of uncertainty from the fire, seismic, and Level 2 PRAs, which are provided in Table B-1, Table C-1, and Table E-1 of the EPRI report. Consistent with Table 3-1 of NUREG-1855, the risk metrics of interest for 10 CFR 50.69 are CDF and LERF. Accordingly, the sources of uncertainty related to Level 2, as listed in Table E-1 of the EPRI report, are evaluated for the PRAs to the extent that they involve the LERF risk metric. Uncertainties related to late releases are not examined because they fall outside of the scope of this investigation.

To determine whether each assumption or uncertainty for the 10 CFR 50.69 application is applicable, the assumption or source of uncertainty was assessed against the criteria listed below. These criteria are based on the definitions in RG 1.200 Revision 2 along with related guidance from NUREG-1855 Revision 1 and related references (i.e., EPRI 1016737, EPRI 1013491, EPRI 1026511, and ASME/ANS RA-Sa-2009). Consistent with Section 4.1 of EPRI Report 1016737 and ASME/ANS RA-Sa-2009, a source of uncertainty is labeled key "when it could impact the PRA results that are being used in a decision, and consequently, may influence the decision being made". EPRI Report 1016737 and ASME/ANS RA-Sa-2009 further indicate that this impact "would need to be significant enough that it changes the degree to which the risk acceptance guidelines are met, and therefore could potentially influence the decision."

Assumptions and sources of uncertainty that do not meet any of the following screening criteria are determined be to potentially key for the application:

1. The uncertainty is addressed by implementing a "consensus model" defined as follows:

   Consensus model – In the most general sense, a consensus model is a model that has a publicly available published basis and has been peer reviewed and widely adopted by an appropriate stakeholder group. In addition, widely accepted PRA practices may be regarded as consensus models. Examples of the latter include the use of the constant probability of failure on demand model for standby components and the Poisson model for initiating events. For risk-informed regulatory decisions, the consensus model approach is one that NRC has utilized or accepted for the specific risk-informed application for which it is proposed. (NUREG-1855).

   Consensus method/model – In the context of risk-informed regulatory decisions, a method or model approach that the NRC has used or accepted for the specific risk-informed application for which it is proposed. A consensus method or model may also have a publicly available, published basis and may have been peer reviewed and widely adopted by an appropriate stakeholder group. (RG 1.200).

   EPRI 1013491 elaborates on the definition of a consensus model to include those areas of the PRA where extensive historical precedent is available to establish a model that has been accepted and yields PRA results that are considered reasonable and realistic. Thus, assumptions for which there is extensive historical precedent, and which produces results that are reasonable and realistic, can be screened from further consideration. According to NRC Regulatory Position C.3.3.2 in RG 1.200. "When a key assumption is shown to be consistent with a consensus method or approach, that key assumption may no longer be subject to additional sensitivity studies in the context of a PRA application."

2. The uncertainty has no impact or insignificant impact on the PRA results and therefore no impact or insignificant impact on the calculated change in risk due to proposed changes that are to be addressed by the PRA application. (EPRI 1016737)

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

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

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

An evaluation of PRA assumptions and generic sources of uncertainty for the 10 CFR 50.69 application was performed. The review determined that none of the PRA uncertainties or assumptions from Appendix A were identified as potential key sources of uncertainty for the 10 CFR 50.69 application.

3.7 Risk Evaluations (10 CFR 50.69(8)(2)(IV))

The SQN 10 CFR 50.69 categorization process will implement the guidance in NEI 00-04 (Reference 3). The overall risk evaluation process described in the NEI guidance addresses both known degradation mechanisms and common cause interactions and meets the requirements of §50.69(b)(2)(iv). Sensitivity studies described in the guidance will be used to confirm that the categorization process results in acceptably small increases to CDF and LERF. The failure rates for equipment and initiating event frequencies used in the PRA include the quantifiable impacts from known degradation mechanisms, as well as other mechanisms (e.g., design errors, manufacturing deficiencies, human errors, etc.). Subsequent performance monitoring and PRA updates required by the rule will continue to capture this data and provide timely insights into the need to account for any important new degradation mechanisms.

The SQN SPRA reflects the current seismic hazard applicable to the plant. TVA will follow industry guidance and common practice in determining whether an update of the SPRA may be warranted due to availability of new consensus seismic hazard information.

3.7.1 Integral Assessment

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

A large majority of SPRA and FPRA basic events are directly aligned with the basic events in other PRA models. The integrated risk importance measures for these components are calculated using the formulae in NEI 00-04 (Section 5.6). However, there are a few SSCs in the SPRA and/or FPRA that are not directly included in the other PRA models.

Subcomponents

The importance of a subcomponent that was not directly modeled in other PRAs will be accounted for in the importance calculation for the component to which it is associated because it can be treated as another failure mode of that component. The decision on the need to treat seismic basic events as representing subcomponents within the importance calculations for another modeled component will be made based on the modeling in each of the PRAs, as part of the PRA basic event-to-component mapping within the categorization process.

SSCs Not in Other PRA Models

There are some SSCs that are unique to the SPRA and/or FPRA. These SSCs may have been screened out of the other PRAs, following the PRA modeling requirements in the ASME/ANS PRA Standard, based on their having no credible failure mode (or an extremely low probability of failure}. If these SSCs are high safety significant (HSS) for the SPRA or FPRA, then their integrated safety significance computation is not necessary. The safety significance would be presented to the Integrated Decision-making Panel (IDP) for their consideration in the decision-making process. The NEI 00-04 process allows the IDP to adjust significance of a SPRA or FPRA modeled SSCs using proper justification. The quantitative integrated importance measure assessment is only one portion of the categorization process.

In summary, most of the seismic and fire basic event importance measures can be directly aligned with components in the Internal Events PRA. Those seismic and/or fire basic events that are not explicitly modeled in the Internal Events PRA, but function as subcomponents of components modeled in the Internal Events PRA, will have their hazard specific importance measures combined with the other PRA importance measures using the NEI 00-04 formulae for the integral assessment. For other seismic and fire basic events that are not explicitly modeled in the Internal Events PRA, an integrated safety significance computation is not necessary because the integrated significance computation is only performed if a SSC modeled in fire or seismic PRA has an initial HSS ranking. The safety significance would be presented to the IDP for their consideration in the decision-making process. The NEI 00-04 process allows the IDP to adjust significance of a SPRA or FPRA modeled SSCs using proper justification.

3.7.2 PRA Uncertainty Evaluations for 10 CFR 50.69 Categorizations

Uncertainty evaluations associated with the risk categorization process are addressed using the processes discussed in the prescribed sensitivity studies discussed in Section 5 of NEI 00-04 (Reference 3).

The SQN 10 CFR 50.69 active categorization process is described in TVA procedures. That process follows the guidance in NEI 00-04 as endorsed in RG 1.201 (Reference 2). Within this process, when a PRA model is used to address the contribution to SSC risk significance due to a given hazard, a set of sensitivity evaluations is required to be performed. TVA procedures describe the active categorization process that encompasses this requirement. Specifically, the recommended set of sensitivity studies to be provided in the FPRA portion of the active categorization are included (as adapted from NEI 00-04, Table 5-3), and the recommended set of sensitivity studies to be provided in the SPRA portion of the active categorization are included (as adapted from NEI 00-04, Table 5-4).

The last item on the list of sensitivities is "Any applicable sensitivity studies identified in the characterization of PRA adequacy and identification of important assumptions and sources of uncertainty." For the current FPRA and SPRA models, no additional PRA-specific sensitivities have been identified that would be expected to have an important impact on categorization results. As the models are updated, the sources of uncertainty will be re-evaluated and, if appropriate, additional sensitivities may be added to the process.

In the overall risk sensitivity studies, TVA utilizes a factor of 3 to increase the unavailability or unreliability of low safety significance (LSS) components. Consistent with the NEI 00-04 guidance (Reference 3), TVA performs both an initial sensitivity study and a cumulative sensitivity study. The initial sensitivity study applies to the system that is being categorized. In the cumulative sensitivity study, the failure probabilities (unreliability and unavailability, as appropriate) of all LSS components modeled in PRAs, including the SPRA and FPRA once this amendment request is approved, for all systems that have been categorized are increased by a factor of 3. This sensitivity study together with the periodic review process assures 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.

3.7.3 Treatment of Seismic Capacity in Uncertainty Evaluations for 10 CFR 50.69 Categorization

TVA proposes to keep the seismic capacity of LSS components as is for the risk sensitivity study outlined in Section 8 of NEI 00-04. This proposal is based on SQN's programs and processes where there is reasonable confidence that the seismic capacities of LSS components would not be impacted by alternative treatment.

TVA has a program for monitoring degradation that could affect the seismic capacity of components at a periodic frequency. The identified degradation is corrected through the standard Condition Reporting and the Corrective Action Program. Should an identified degradation appear to challenge a SPRA modeling aspect, then an impact evaluation on the results of the SPRA would be performed to determine if the original categorization remains valid. Thus, the monitoring program for SSCs ensures that potential degradation of the seismic capacity would be detected and addressed before significantly impacting the seismic risk. TVA

has implemented a rigorous configuration management program to maintain the configuration of SSCs in the plant. Unless an item to be procured is equivalent to an existing item (e.g., like-for-like replacement), an appropriate design change process is utilized to ensure that design requirements remain unchanged as required by the 10 CFR 50.69 rule. In addition, as stated in the 10 CFR 50.69 rule, RISC-3 SSCs must meet the following requirements: (1) meet fracture toughness requirements for Class 2 and 3 components and (2) remain capable of performing their safety-related functions under design basis conditions, including seismic conditions. The procurement activities are developed and implemented to meet the above requirements.
In summary, based on the SQN 10 CFR 50.69 program procedures, and the supporting plant procedures, there is reasonable confidence that the seismic capacities of LSS components would not be impacted such that the plant CDF or LERF would be significantly affected. Thus, an inclusion of LSS components in a sensitivity study required by NEI section 8.0 is not warranted to evaluate seismic capacity.

4.0 REGULATORY ANALYSIS

4.1 Applicable Regulatory Requirements and Criteria

The following NRC requirements and guidance documents are applicable to the proposed change.

- The regulations at Title 10 of the *Code of Federal Regulations* (CFR) Part 50.69, "Risk-Informed Categorization and Treatment of Structures, Systems and Components for Nuclear Power Reactors."
- NRC Regulatory Guide 1.201, "Guidelines for Categorizing Structures, Systems, and Components in Nuclear Power Plants According to their Safety Significance," Revision 1, May 2006.
- Regulatory Guide 1.174, "An Approach for Using Probabilistic Risk Assessment in Risk-Informed Decisions on Plant-Specific Changes to the Licensing Basis," Revision 2, April 2015.
- Regulatory Guide 1.200, "An Approach for Determining the Technical Adequacy of Probabilistic Risk Assessment Results for Risk-Informed Activities," Revision 2, March 2009.

4.2 Precedent

This license amendment request cites the precedent of NRC approval of the Southern Nuclear Operating Company, Inc., application for the Vogtle Electric Generating Plant, Units 1 and 2 regarding the application of seismic probabilistic risk assessment into the previously approved 10 CFR 50.69 categorization process on August 10, 2018 (Reference 19).

4.3 No Significant Hazards Consideration

TVA proposes to modify the licensing basis to amend the approved voluntary implementation of the provisions of 10 CFR, Part 50.69, "Risk-Informed Categorization and Treatment of Structures, Systems, and Components (SSC) for Nuclear Power Plants" to include use of the SQN SPRA in place of the Seismic Safe Shutdown Equipment List from the seismic margin analysis for seismic risk, and additionally the use of the SQN FPRA in place of the fire safe shutdown equipment list based on the 10 CFR 50 Appendix R analysis for fire risk. The provisions of 10 CFR 50.69 allow adjustment of the scope of equipment subject to special treatment controls (e.g., quality assurance, testing, inspection, condition monitoring, assessment, and evaluation). For equipment determined to be of low safety significance, alternative treatment requirements can be implemented in accordance with this regulation. For equipment determined to be of high safety significance, requirements will either not be changed, or will be enhanced. This allows improved focus on equipment that has safety significance resulting in improved plant safety.

As required by 10 CFR 50.91(a), an analysis of the issue of no significant hazards consideration is presented below:

1. *Does the proposed change involve a significant increase in the probability or consequences of an accident previously evaluated?*

   Response: No.

   The proposed change replaces the use of the SQN SMA with use of the peer reviewed SQN SPRA, and the use of the alternate method of internal fire with use of the peer reviewed SQN FPRA, within the NRC-approved risk-informed categorization process to modify the scope of SSCs subject to NRC special treatment requirements and to implement alternative treatments per the regulations. The use of an SPRA in place of an SMA, and FPRA in place of the alternate method of internal fire, is allowed by the 10 CFR 50.69 process guidance defined in NEI 00-04, and as endorsed by NRC in RG 1.201. The process used to evaluate SSCs for changes to NRC special treatment requirements and the use of alternative requirements ensures the ability of the SSCs to perform their design function. The potential change to special treatment requirements does not change the design and operation of the SSCs. As a result, the proposed change does not affect any initiators to accidents previously evaluated or the ability to mitigate any accidents previously evaluated. The consequences of the accidents previously evaluated are not affected because the mitigation functions performed by the SSCs assumed in the safety analysis are not being modified. The SSCs required to safely shut down the reactor and maintain it in a safe shutdown condition following an accident will continue to perform their design functions.

   Therefore, the proposed change does not involve a significant increase in the probability or consequences of an accident previously evaluated.

2. *Does the proposed change create the possibility of a new or different kind of accident from any accident previously evaluated?*

   Response: No.

   The proposed change continues to permit the use of a risk-informed categorization process to modify the scope of SSCs subject to NRC special treatment requirements and to implement alternative treatments per the regulations. The proposed change does not change the functional requirements, configuration, or method of operation of any SSC. Under the proposed change, no additional plant equipment will be installed.

   Therefore, the proposed change does not create the possibility of a new or different kind of accident from any accident previously evaluated.

3. *Does the proposed change involve a significant reduction in a margin of safety?*

   Response: No.

   The proposed change will continue to permit the use of a risk-informed categorization process to modify the scope of SSCs subject to NRC special treatment requirements and to implement alternative treatments per the regulations. The proposed change does not affect any safety limits or operating parameters used to establish the safety margin. The safety margins included in analyses of accidents are not affected by the proposed change. The regulation requires that there be no significant effect on plant risk due to any change to the special treatment requirements for SSCs and that the SSCs continue to be capable of performing their design basis functions, as well as to perform any beyond design basis functions consistent with the categorization process and results.

   Therefore, the proposed change does not involve a significant reduction in a margin of safety.

Based on the responses above, TVA concludes that the proposed change presents no significant hazards consideration under the standards set forth in 10 CFR 50.92(c), and, accordingly, a finding of "no significant hazards consideration" is justified.

4.4 Conclusion

In conclusion, based on the considerations discussed above, (1) there is reasonable assurance that the health and safety of the public will not be endangered by operation in the proposed manner, (2) such activities will be conducted in compliance with NRC regulations, and (3) the issuance of the amendment will not be inimical to the common defense and security or to the health and safety of the public.

5.0 ENVIRONMENTAL CONSIDERATION

A review has determined that the proposed amendment would change a requirement with respect to installation or use of a facility component located within the restricted area, as defined in 10 CFR 20, or would change an inspection or surveillance requirement. However, the proposed amendment does not involve (i) a significant hazards consideration, (ii) a significant change in the types or significant increase in the amounts of any effluents that may be released offsite, or (iii) a significant increase in individual or cumulative occupational radiation exposure. Accordingly, the proposed amendment meets the eligibility criterion for categorical exclusion set forth in 10 CFR 51.22(c)(9). Therefore, pursuant to 10 CFR 51.22(b), no environmental impact statement or environmental assessment need be prepared in connection with the proposed amendment.

6.0 REFERENCES

1. NRC Letter to TVA, “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 (ML19179A135)

2. NRC Regulatory Guide 1.201, “Guidelines for Categorizing Structures, Systems, and Components in Nuclear Power Plants According to their Safety Significance,” Revision 1, May 2006

3. NEI 00-04, “10 CFR 50.69 SSC Categorization Guideline,” Revision 0, Nuclear Energy Institute, July 2005

4. TVA Letter to NRC, CNL-17-010, “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),” dated March 16, 2018 (ML18075A365)

5. Generic Letter 88-20, “Individual Plant Examination of External Events (IPEEE) for Severe Accident Vulnerabilities - 10 CFR 50.54(f), Supplement 4,” U.S. NRC, June 1991

6. NUREG-2115, Central and Eastern United States Seismic Source Characterization for Nuclear Facilities, January 2012

7. EPRI (2015), Central and Eastern United States Seismic Source Characterization for Nuclear Facilities; Maximum Magnitude Distribution Evaluation, EPRI, Palo Alto, CA 2015, Report Number 3002005684, June 2015

8. EPRI (2013a) EPRI (2004, 2006) Ground-Motion Model (GMM) Review Project, EPRI Report Number 3002000717, June 2013

9. EPRI (2013b), Errata Sheet for EPRI Technical Report EPRI (2004, 2006) Ground-Motion Model (GMM) Review Project, Product 3002000717, August 15, 2013

10. NUREG/CR-6372, Recommendations for Probabilistic Seismic Hazard Analysis: Guidance on Uncertainty and Use of Experts, April 1997

11. NUREG-2117, Practical Implementation Guidelines for SSHAC Level 3 and 4 Hazard Studies, Revision 1, April 2012

12. ASME/ANS RA-Sb-2013, “Standard for Level I/Large Early Release Frequency Probabilistic Risk Assessment for Nuclear Power Plant Applications, Addendum B to RA-S-2008,” July 2013

13. NEI 12-13, “External Hazards PRA Peer Review Process Guidelines,” Revision 3, August 2012

14. NRC Letter to NEI, “U.S. Nuclear Regulatory Commission Acceptance of Nuclear Energy Institute (NEI) Guidance NEI-12-13, ‘External Hazards PRA Peer Review Process Guidelines’,” (August 2012), March 7, 2018 (ML18025C024 and ML18025C025)

15. APC 17-13, Subject: "NRC Acceptance of Industry Guidance on Closure of PRA Peer Review Findings," dated May 8, 2017, with attachment Appendix X

16. NEI letter to NRC, “Final Revision of Appendix X to NEI 05-04/07-12/12-16, *Close-Out of Facts and Observations (F&Os)*,” February 21, 2017 (ML17086A431)

17. NRC letter to NEI, “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),” May 3, 2017 (ML17079A427)

18. Southern Nuclear letter to NRC, NL-17-1201, "Vogtle Electric Generating Plant Units 1 & 2 Response to Supplemental Information Needed for Acceptance of Systematic Risk-Informed Assessment of Debris Technical Report," dated July 11, 2017 (ML17192A245)

19. NRC letter to Southern Nuclear Operating Company, Inc., "Vogtle Electric Generating Plant, Units 1 and 2 – Issuance of Amendments Regarding Application of Seismic Probabilistic Risk Assessment into the Previously Approved 10 CFR 50.69 Categorization Process (EPID L-2017-LLA-0248)," dated August 10, 2018 (ML18180A062)

20. ASME/ANS RA-Sa-2009, "Standard for Level I/Large Early Release Frequency Probabilistic Risk Assessment for Nuclear Power Plant Applications, Addendum A to RA-S-2008," February 2009

21. NRC Regulatory Guide 1.200, "An Approach for Determining the Technical Acceptability of Probabilistic Risk Assessment Results for Risk-Informed Activities," Revision 2, March 2009 (ML090410014)

22. NEI Topical Report NEI 07-12, "Fire Probabilistic Risk Assessment (FPRA) Peer Review Process Guidelines," Revision 1, June 2010 (ML102230070)

23. NUREG-1792, "Good Practices for Implementing Human Reliability Analysis (HRA), Sandia National Laboratories," April 2005

24. NUREG-1855, "Guidance on the Treatment of Uncertainties Associated with PRAs in Risk-Informed Decisionmaking," Revision 1, March 2017

Attachment 1

Current Applicable SPRA and FPRA Models

The purpose of this attachment is to demonstrate that the Tennessee Valley Authority (TVA) Sequoyah Nuclear Plant (SQN) Plant total core damage frequency (CDF) and total large early release frequency (LERF) are below the guidelines established in Regulatory Guide (RG) 1.174. RG 1.174 does not establish firm limits for total CDF and LERF but recommends that risk-informed applications be implemented only when the total plant risk is no more than 1E-4/year and 1E-5/year, respectively. Demonstrating that these limits are met confirms that the risk metrics of Nuclear Energy Institute (NEI) 00-04 can be applied to the SQN Title 10 of the *Code of Federal Regulations* (CFR) 50.69 Risk Categorization Program.

The TVA SQN Probabilistic Risk Assessment (PRA) model update process includes "model of record" updates which are full scope model updates that include all documentation required by the American Society of Mechanical Engineers /American Nuclear Society (ASME/ANS) RA-Sa-2009 PRA Standard (hereafter "ASME/ANS PRA Standard"), "Addenda to ASME/ANS RA-S-2008 Standard for Level 1/Large Early Release Frequency Probabilistic Risk Assessment for Nuclear Power Plant Applications," as endorsed in RG 1.200. The update process and the associated procedures implement plant changes, or correct errors to support one or more risk-informed applications. The current internal events including internal flooding PRA model of record (MOR) is MOR 5, the FPRA is MOR 2 and the SPRA is MOR 2.

The 10 CFR 50.69 risk metrics are calculated by using a One-Top Multi-Hazard Model (OTMHM). The OTMHM was created by merging the one-top models for Internal Events and Internal Flooding PRA (MOR 5), SPRA (MOR 2), and FPRA (MOR 2) into a single fault tree. This model was optimized to increase the quantification speed. The optimized OTMHM was quantified (by individual hazard) to demonstrate that the TVA SQN Plant total CDF and total LERF are below the guidelines established in RG 1.174.

Table 1 lists the Unit 1 and Unit 2 CDF and LERF values as calculated by the optimized OTMHM. Other external hazards do not contribute significantly to the total risk.

Table 1. Total Baseline Model of Record CDF/LERF from OTMHM

SQN Unit 1 Baseline CDF

Source	Contribution
Internal Events PRA (Including Flooding)	4.88E-06
Fire PRA	6.21E-05
Seismic CDF	4.19E-06
Other External Events	No significant contribution
Total Unit 1 CDF	**7.12E-05**

SQN Unit 1 Baseline LERF

Source	Contribution
Internal Events PRA (Including Flooding)	6.63E-07
Fire PRA	4.35E-06
Seismic LERF	3.00E-06
Other External Events	No significant contribution
Total Unit 1 LERF	**8.01E-06**

SQN Unit 2 Baseline CDF

Source	Contribution
Internal Events PRA (Including Flooding)	5.19E-06
Fire PRA	6.63E-05
Seismic CDF	3.95E-06
Other External Events	No significant contribution
Total Unit 2 CDF	**7.54E-05**

SQN Unit 2 Baseline LERF

Source	Contribution
Internal Events PRA (Including Flooding)	6.97E-07
Fire PRA	5.32E-06
Seismic LERF	2.83E-06
Other External Events	No significant contribution
Total Unit 2 LERF	**8.85E-06**

Notes:

1. Results presented in this table were calculated using the optimized OTMHM.
2. Other external hazards have been screened as discussed in the Enclosure of this license amendment request.

Attachment 2

Proposed Operating License Changes (Units 1 and 2 Markups)
(3 total pages)

(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.

a fire PRA; a seismic PRA

(as revised by License Amendment ___)

(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 NUMARC 96-01 shutdown safety assessment process to assess shutdown risk to a shutdown PRA approach 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. 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:

(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"

a fire PRA; a seismic PRA

(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~~.~~ (as revised by License Amendment ___)

(2) Prior to implementation of the provisions of 10 CFR 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 the NUMARC 96-01 shutdown safety assessment process to assess shutdown risk to a shutdown PRA approach ~~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~~).

(28) Prior to Cycle 24 startup from Unit 2 Refueling Outage 23, TVA shall ensure the Cycle 24 core design will not adversely affect the safety of the plant in accordance with TVA procedure, NFDP-111, “Nuclear Design and Core Analysis.”

Attachment 3

Proposed Operating License Pages (Units 1 and 2 Final Retyped)
(3 total pages)

(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; a fire PRA; a seismic PRA; 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., a screening of 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 (as revised by License Amendment ___).

(2) Prior to implementation of the provisions of 10 CFR 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 the NUMARC 96-01 shutdown safety assessment process to assess shutdown risk to a shutdown PRA approach).

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>

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(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) DELETED

(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; a fire PRA; a seismic PRA; 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., a screening of external hazards updated using the criteria in the endorsed ASME/ANS RA-Sa-2009 PRA Standard for external hazard screening significance; as specified in Unit 2 License Amendment 340 (as revised by License Amendment ___).

(2) Prior to implementation of the provisions of 10 CFR 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 the NUMARC 96-01 shutdown safety assessment process to assess shutdown risk to a shutdown PRA approach).

(28) Prior to Cycle 24 startup from Unit 2 Refueling Outage 23, TVA shall ensure the Cycle 24 core design will not adversely affect the safety of the plant in acordance with TVA procedure, NFDP-111, “Nuclear Design and Core Analysis.”