Tennessee Valley Authority, 1101 Market Street, Chattanooga, Tennessee 37402

CNL-17-010

March 16, 2018

10 CFR 50.69
10 CFR 50.90

ATTN: Document Control Desk
U. S. Nuclear Regulatory Commission
Washington, DC 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 328

SUBJECT: **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)**

References:
1. TVA Letter to NRC, "Sequoyah Nuclear Plants Units 1 and 2 Technical Specifications Conversion to NUREG 1431, Revision 4.0 (SQN-TS-11-10)," dated November 22, 2013 (ML 13329A881)
2. NRC Letter to TVA, "Sequoyah Nuclear Plant, Units 1 and 2 - Issuance of Amendments for the Conversion to the Improved Technical Specifications with Beyond Scope Issues (TAC Nos. MF3128 and MF3129)," dated September 30, 2015 (ML15238B460)
3. Technical Specification Task Force (TSTF)-425, Revision 3, "Relocate Surveillance Frequencies to Licensee Control - RITSTF Initiative 5b," dated March 18, 2009 (ML090850627)
4. TSTF-446 "Risk Informed Evaluation of Extensions to Containment Isolation Valve Completion Times (WCAP-15791),"Revision 3, dated February 19, 2008 (ML080510164)

In accordance with the provisions of Title 10 of the *Code of Federal Regulations* (10 CFR) 50.69 and 10 CFR 50.90, Tennessee Valley Authority (TVA) is submitting for Nuclear Regulatory Commission (NRC) approval, a request for an amendment to Renewed Facility Operating License Nos. DPR-77 and DPR-79 for the Sequoyah Nuclear Plant (SQN), Units 1 and 2.

The proposed license amendment request (LAR) would modify the SQN Renewed Facility Operating Licenses to allow for the implementation of the provisions of 10 CFR, Part 50.69, "Risk-Informed Categorization and Treatment of Structures, Systems and Components for Nuclear Power Reactors." The provisions of 10 CFR 50.69 allow adjustment of the scope of 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 not be changed or will be enhanced. This allows improved focus on equipment that has safety significance resulting in improved plant safety.

Enclosure 1 to this letter provides the basis for the proposed change to the SQN, Units 1 and 2 Renewed Facility Operating Licenses. The categorization process being implemented through this change is consistent with NEI 00-04, "10 CFR 50.69 SSC Categorization Guideline," Revision 0, dated July 2005, which was endorsed by the NRC in Regulatory Guide 1.201, "Guidelines for Categorizing Structures, Systems, and Components in Nuclear Power Plants According to their Safety Significance," Revision 1 dated May 2006. Attachment 1 of the enclosure provides a list of categorization prerequisites. Use of the categorization process on a plant system would only occur after these prerequisites are met.

The NRC has previously reviewed the technical adequacy of the SQN Probabilistic Risk Assessment (PRA) model identified in this application during the TVA application to convert the SQN TS to Improved Technical Specifications (ITS) (Reference 1). The NRC approval and associated Safety Evaluations for SQN's application of ITS were provided in Reference 2. TSTF-425 and TSTF-446 (References 3 and 4 respectively), were included in the SQN ITS application (Reference 1). The risk analysis described in this 10 CFR 50.69 application utilizes the same PRA models described in the ITS application (Reference 1).

TVA included these risk-informed changes in the application to convert to ITS (Reference 1). Therefore, the PRA Technical Adequacy was reviewed for these changes simultaneously.

TVA requests that the NRC utilize the review of the PRA Technical Adequacy included in the ITS application when performing the review for compliance with 10 CFR 50.69(b)(2)(ii) and 50.69(b)(2)(iii) in this LAR.

Enclosure 2 provides the existing SQN Unit 1 and Unit 2 Renewed Operating Licenses marked-up to show the proposed changes. Enclosure 3 provides the existing SQN Unit 1 and Unit 2 Renewed Operating Licenses pages to show the proposed changes.

TVA has determined that there are no significant hazards consideration associated with the proposed change and that the license amendment qualifies for a categorical exclusion from environmental review pursuant to the provisions of 10 CFR 51.22(c)(9). Additionally, in accordance with 10 CFR 50.91(b)(1), TVA is sending a copy of this letter and enclosures to the Tennessee Department of Environment and Conservation.

TVA requests approval of the proposed license amendment within 12 months of the date of this letter with implementation within 60 days following NRC approval.

There are no new regulatory commitments made in this letter. Section 2.3 of Enclosure 1 describes the proposed license condition associated with this LAR. Please address any questions regarding this submittal to Edward D. Schrull at (423) 751-3850

I declare under penalty of perjury that the foregoing is true and correct. Executed on the 16th day of March 2018.

Respectfully,

for

J. W. Shea
Vice President, Nuclear Regulatory Affairs and Support Services

Enclosures:

1. Evaluation of the Proposed Change
2. Proposed Changes (Mark-Ups) to SQN Renewed Operating Licenses for Unit 1 and Unit 2
3. Proposed Changes (Final Typed) to SQN Renewed Operating Licenses for Unit 1 and Unit 2

cc: NRC Regional Administrator - Region II
NRC Senior Resident Inspector - Sequoyah Nuclear Plant
NRC Project Manager - Sequoyah Nuclear Plant
Division of Radiological Health - Tennessee State Department of Environment and Conservation

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LIST OF ATTACHMENTS

1. SUMMARY DESCRIPTION

The proposed amendment modifies the Tennessee Valley Authority (TVA) Sequoyah Nuclear Plant (SQN), Units 1 and 2 licensing basis to allow for the 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 for Nuclear Power Reactors.” 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 not be changed or will be enhanced. This allows improved focus on equipment that has safety significance resulting in improved plant safety.

2. DETAILED DESCRIPTION

2.1 CURRENT REGULATORY REQUIREMENTS

The Nuclear Regulatory Commission (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 (DBEs). 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. The structures, systems, and components (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 and high reliability, and have the capability to perform during postulated design basis conditions. 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," is not defined in the General Design Criteria (GDC) of Appendix A to 10 CFR Part 50.

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, providing a logical means for prioritizing these challenges based on safety significance, and allowing consideration of a broader set of resources to defend against these challenges. In contrast to the deterministic approach, Probabilistic Risk Assessments (PRAs) address 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, alternative treatment requirements can be implemented in accordance with the regulation. For equipment determined to be of high safety significance, requirements will not be changed or will be enhanced. This allows improved focus on equipment that has safety significance 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 class (RISC) categories. The determination of safety significance is performed by an integrated decision-making process, as described by NEI 00-04, “10 CFR 50.69 SSC Categorization Guideline,” (Reference 1), which uses both risk insights and traditional engineering insights. The safety functions include the design basis functions, as well as functions credited for severe accidents (including external events). Special or alternative treatment for the SSCs is applied as necessary to maintain functionality and reliability, and is a function of the SSC categorization results and associated bases. Finally, periodic assessment activities are conducted to make adjustments to the categorization and/or 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. For SSCs that are categorized as high safety significant (HSS), existing treatment requirements are maintained or enhanced. Conversely, for SSCs that do not significantly contribute to plant safety on an individual basis, the rule allows an alternative risk-informed approach to treatment that provides reasonable, though reduced, level of confidence that these SSCs will satisfy functional requirements.

Implementation of 10 CFR 50.69 will allow TVA to improve focus on equipment that has safety significance resulting in improved plant safety.

2.3 DESCRIPTION OF THE PROPOSED CHANGE

TVA proposes the addition of the following condition to the Renewed Operating Licenses of SQN, Units 1 and 2 to document the NRC's approval of the use of 10 CFR 50.69.

> *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) specified in the license amendment dated [XXXX].*
>
> *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).*
>
> *TVA shall complete the items listed in Attachment 1, List of Categorization Prerequisites, of TVA letter dated [XXXX], prior to implementation.*

3. TECHNICAL EVALUATION

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 § 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 § 50.69(c)(1)(i).
>
> (iv) A description of, and basis for acceptability of, the evaluations to be conducted to satisfy § 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).

Each of these submittal requirements are addressed in the following sections of this enclosure.

The NRC has previously reviewed the technical adequacy of the SQN PRA model identified in this application during the TVA application to convert the SQN Technical Specifications (TS) to Improved Technical Specifications (ITS) (Reference 2). The NRC approval and associated Safety Evaluations for Sequoyah's application of ITS were provided in Reference 3. Technical Specification Task Force (TSTF)-425 and TSTF-446 (References 4 and 5, respectively), were included in the TVA ITS application (Reference 2). The risk analysis described in this 10 CFR 50.69 application utilizes the same PRA models described in the ITS application (Reference 2).

TVA requests that the NRC utilize the review of the PRA Technical Adequacy for the ITS application when performing the review for compliance with 10 CFR 50.69(b)(2)(ii) and 50.69(b)(2)(iii) in this license amendment request (LAR).

3.1 CATEGORIZATION PROCESS DESCRIPTION (10 CFR 50.69(b)(2)(I))

3.1.1 Overall Categorization Process

TVA will implement the risk categorization process in accordance with the NEI 00-04, Revision 0, as endorsed by Regulatory Guide (RG) 1.201, "Guidelines for Categorizing Structures, Systems, and Components in Nuclear Power Plants According to their Safety Significance" (Reference 6). NEI 00-04 Section 1.5 states, "Due to the varying levels of uncertainty and degrees of conservatism in the spectrum of risk contributors, the risk significance of SSCs is assessed separately from each of five risk perspectives and used to identify SSCs that are potentially safety-significant." A separate evaluation is appropriate to avoid reliance on a combined result that may mask the results of individual risk contributors.

The process to categorize each system will be consistent with the guidance in NEI 00-04, "10 CFR 50.69 SSC Categorization Guideline," as endorsed by RG 1.201. RG 1.201 states that "the implementation of all processes described in NEI 00-04 (i.e., Sections 2 through 12) is integral to providing reasonable confidence" and that "all aspects of NEI 00-04 must be followed to achieve reasonable confidence in the evaluations required by §50.69(c)(1)(iv)." However, neither RG 1.201 nor NEI 00-04 prescribe a particular sequence or order for each of the elements to be completed. Therefore, the order in which each of the elements of the categorization process (listed below) is completed is flexible and as long as they are all completed they may even be performed in parallel. Note that NEI 00-04 only requires Item 3 to be completed for components/functions categorized as LSS by all other elements. Similarly, NEI 00-04 only requires Item 4 to be completed for safety related active components/functions categorized as LSS by all other elements.

1. PRA-based evaluations (e.g., the internal events, internal flooding, and fire PRAs).
2. non-PRA approaches (e.g., fire safe shutdown equipment list (SSEL), seismic safe shutdown equipment list (SSEL), other external events screening, and shutdown assessment).
3. Seven qualitative criteria in Section 9.2 of NEI 00-04.
4. the defense-in-depth assessment.
5. the passive categorization methodology

Categorization of SSCs will be completed per the NEI 00-04 process, as endorsed by RG 1.201, which includes the determination of safety significance through the various elements identified above. The results of these elements are used as inputs to arrive at a preliminary component categorization (i.e., High Safety Significant (HSS) or Low Safety Significant (LSS)) that is presented to the Integrated Decision-Making Panel (IDP). Note: the term “preliminary HSS or LSS” is synonymous with the NEI 00-04 term “candidate HSS or LSS.” A component or function is preliminarily categorized as HSS if any element of the process results in a preliminary HSS determination in accordance with Table 3-1 below. The safety significance determination of each element, identified above, is independent of each other and therefore the sequence of the elements does not impact the resulting preliminary categorization of each component or function. Consistent with NEI 00-04, the categorization of a component or function will only be “preliminary” until it has been confirmed by the IDP. Once the IDP confirms that the categorization process was followed appropriately, the final Risk Informed Safety Class (RISC) category can be assigned.

The IDP may direct and approve detailed categorization of components in accordance with NEI 00-04 Section 10.2. The IDP may always elect to change a preliminary LSS component or function to HSS, however the ability to change component categorization from preliminary HSS to LSS is limited. This ability is only available to the IDP for select process steps as described in NEI 00-04 and endorsed by RG 1.201. Table 3-1 summarizes these IDP limitations in NEI 00-04. The steps of the process are performed at either the function level, component level, or both. This is also summarized in the Table 3-1. A component is assigned its final RISC category upon approval by the IDP.

Table 3-1: IDP Changes from Preliminary HSS to LSS

Element	Categorization Step - NEI 00-04 Section	Evaluation Level	IDP Change HSS to LSS	Drives Associated Functions
Risk (PRA Modeled)	Internal Events Base Case – Section 5.1	Component	Not Allowed	Yes
	Fire, Seismic and Other External Events Base Case		Allowable	No
	PRA Sensitivity Studies		Allowable	No
	Integral PRA Assessment – Section 5.6		Not Allowed	Yes
Risk (Non-modeled)	Fire, Seismic and Other External Hazards –	Component	Not Allowed	No
	Shutdown – Section 5.5	Function/Component	Not Allowed	No
Defense-in-Depth	Core Damage – Section 6.1	Function/Component	Not Allowed	Yes
	Containment – Section 6.2	Component	Not Allowed	Yes
Qualitative Criteria	Considerations – Section 9.2	Function	Allowable	N/A
Passive	Passive – Section 4	Segment/Component	Not Allowed	No

The mapping of components to system functions is used in some categorization process steps to facilitate preliminary categorization of components. Specifically, functions with mapped components that are determined to be HSS by the PRA-based assessment (i.e., Internal events PRA or Integral PRA assessment) or defense-in-depth evaluation will be initially treated as HSS. However, NEI 00-04 Section 10.2 allows detailed categorization which can result in some components mapped to HSS functions being treated as LSS; and Section 4.0 discusses additional functions that may be identified (e.g., fill and drain) to group and consider potentially LSS components that may have been initially associated with a HSS function but which do not support the critical attributes of that HSS function. Note that certain steps of the categorization process are performed at a component level (e.g. Passive, Non-PRA-modeled hazards - see Table 3-1). These components from the component level assessments will remain HSS (IDP cannot override) regardless of the significance of the functions to which they are mapped. Therefore, if a HSS component is mapped to a LSS function, that component will remain HSS. If an LSS component is mapped to an HSS function, that component may be driven HSS based on Table 3-1 above, or may remain LSS.

The following are clarifications to be applied to the NEI 00-04 categorization process:

- The Integrated Decision-Making Panel (IDP) will be composed of a group of at least five experts who collectively have expertise in plant operation, design (mechanical and electrical) engineering, system engineering, safety analysis, and probabilistic risk assessment. At least three members of the IDP will have a minimum of five years of experience at the plant, and there will be at least one member of the IDP who has a minimum of three years of experience in the modeling and updating of the plant-specific PRA.

- The IDP will be trained in the specific technical aspects and requirements related to the categorization process. Training will address at a minimum the purpose of the categorization; present treatment requirements for SSCs including requirements for design basis events; PRA fundamentals; details of the plant specific PRA including the modeling, scope, and assumptions, the interpretation of risk importance measures, and the role of sensitivity studies and the change-in-risk evaluations; and the defense-in-depth philosophy and requirements to maintain this philosophy.

- The decision criteria for the IDP for categorizing SSCs as safety significant or low safety-significant pursuant to § 50.69(f)(1) will be documented in TVA procedures. Decisions of the IDP will be arrived at by consensus. Differing opinions will be documented and resolved, if possible. If a resolution cannot be achieved concerning the safety significance of an SSC, then the SSC will be classified as safety-significant.

- Passive characterization will be performed using the processes described in Section 3.1.2. Consistent with NEI 00-04, an HSS determination by the passive categorization process cannot be changed by the IDP.

- An unreliability factor of 3 will be used for the sensitivity studies described in Section 8 of NEI 00-04. The factor of 3 was chosen as it is representative of the typical error factor of basic events used in the PRA model.

- NEI 00-04 Section 7 requires assigning the safety significance of functions to be preliminary HSS if it is supported by an SSC determined to be HSS from the PRA-based assessment in Section 5, but does not require this for SSCs determined to be HSS from non-PRA-based, deterministic assessments in Section 5. This requirement is further clarified in the Vogtle SER (Reference X) which states "…if any SSC is identified as HSS from either the integrated PRA component safety significance assessment (Section 5 of NEI 00-04) or the defense-in-depth assessment (Section 6), the associated system function(s) would be identified as HSS."

- Once a system function is identified as HSS, then all the components that support that function are preliminary HSS. The IDP must intervene to assign any of these HSS Function components to low safety significant (LSS).

- With regard to the criterion that considers whether the active function is called out or relied upon in the plant Emergency/Abnormal Operating Procedures, TVA will not take credit for alternate means unless the alternate means are proceduralized and included in Licensed Operator training.

The following exception is taken to the NEI 00-04 categorization process:

- NEI 00-04, Section 5.2 states that the fire safety significance process takes one of two forms. Either the use of Fire Induced Vulnerability Evaluation (FIVE) or a Fire PRA. However, Section 3.2.2 of this LAR describes an alternate approach, which implements the Appendix R Safe Shutdown analysis that will be used in the SQN categorization process to evaluate safety significance related to the fire hazard.

The risk analysis to be implemented for each hazard is described below.

- Internal Event Risks: Internal Events including internal flooding PRA model Revision 3, dated August 5, 2014 or later updated model as described in section 3.2.6 of this enclosure will be used. This model was accepted by NRC in Reference 3.

- Internal Fire Risks: The SQN Appendix R Safe Shutdown Equipment List (SSEL) will be used. The SQN Appendix R SSEL is a living program that reflects the current as-built, as-operated plant.

- Seismic Risks: The SSEL from the Individual Plant Examination of External Events (IPEEE) Seismic Margins Analysis (SMA) approved by the NRC in Reference 7 will be used.

- Other External Risks (e.g., tornados, external floods): The IPEEE screening process approved by the NRC in Reference 7 will be used. The other external hazards were determined to be insignificant contributors to plant risk.

- Low Power and Shutdown Risks: The Qualitative defense-in-depth (DID) shutdown model for shutdown configuration risk management (CRM) based on the framework for DID provided in Reference 8, which provides guidance for assessing and enhancing safety during shutdown operations.

A change to the categorization process that is outside the bounds specified above (e.g., change from a seismic margins approach to a seismic probabilistic risk assessment approach) will not be used without prior NRC approval. The SSC categorization process documentation will include the following elements.

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

3.1.2 Passive Categorization Process

For the purposes of 10 CFR 50.69 categorization, passive components are those components that have a pressure retaining function. Passive components and the passive function of active components will be evaluated using the Arkansas Nuclear One (ANO) Risk-Informed Repair/Replacement Activities (RI-RRA) methodology contained in Reference 9 consistent with the related Safety Evaluation Report (SER) issued by the Office of Nuclear Reactor Regulation.

The RI-RRA methodology is a risk-informed safety classification and treatment program for repair/replacement activities (RI-RRA methodology) for pressure retaining items and their associated supports. In this method, the component failure is assumed with a probability of 1.0 and only the consequence evaluation is performed. It additionally applies deterministic considerations (e.g., DID, safety margins) in determining safety significance. Component supports are assigned the same safety significance as the highest passively ranked component within the bounds of the associated analytical pipe stress model. Consistent with NEI 00-04, an HSS determination by the passive categorization process cannot be changed by the IDP.

The use of this method was previously approved to be used for a 10 CFR 50.69 application by NRC in the final Safety Evaluation for Vogtle dated December 17, 2014 (Reference 10). The RI-RRA method as approved for use at Vogtle for 10 CFR 50.69 does not have any plant specific aspects and is generic. It relies on the conditional core damage and large early release probabilities associated with postulated ruptures. Safety significance is generally measured by the frequency and the consequence of the event. However, this RI-RRA process categorizes components solely based on consequence, which measures the safety significance of the passive component given that it ruptures. This approach is conservative compared to including the rupture frequency in the categorization as this approach will not allow the categorization of SSCs to be affected by any changes in frequency due to changes in treatment. The

passive categorization process is intended to apply the same risk-informed process accepted in the ANO2-R&R-004 for the passive categorization of Class 2, 3, and non-class components. Consistent with ANO2-R&R-004, Class 1 pressure retaining SSCs in the scope of the system being categorized will be assigned HSS and cannot be changed by the IDP. Therefore, the RI-RRA methodology for passive categorization is acceptable and appropriate for use at SQN for 10 CFR 50.69 SSC Categorization.

3.2 TECHNICAL ADEQUACY EVALUATION (10 CFR 50.69(b)(2)(II))

The following sections demonstrate that the quality and level of detail of the processes used in categorization of SSCs are adequate. The PRA models described below have been peer reviewed; there are no PRA upgrades that have not been peer reviewed. The PRA models credited in this request are the same PRA models credited in Reference 3, with changes to the as-built, as-operated plant considered with respect to impact on the model that could potentially require an interim model update.

3.2.1 Internal Events and Internal Flooding

The SQN categorization process for the internal events and flooding hazard will use the plant-specific PRA model. The TVA risk management process ensures that the PRA model used in this application reflects the as-built and as-operated plant for each of the SQN units. Attachment 2 of this enclosure identifies the applicable internal events and internal flooding PRA models.

3.2.2 Fire Hazards

The SQN categorization process will use the Fire SSEL for evaluation of safety significance related to fire hazards. The Fire Safe Shutdown paths identify the safety functions and associated sets of equipment credited to achieve and maintain safe shutdown under postulated fire conditions as defined by 10 CFR 50, Appendix R and NRC Branch Technical Position CMEB 9.5-1 (Reference 11). The Fire SSEL identifies the credited equipment on these Fire Safe Shutdown Paths. The approach also considers regulatory exemptions related to the Fire Safe Shutdown program and fire-induced Multiple Spurious Operations (MSOs) to identify any additional equipment.

The use of the Fire SSEL is a screening approach. There are no importance measures used in determining safety significance related to the fire hazard. Instead, using the Fire SSEL would identify all credited equipment as HSS regardless of their fire damage susceptibility or frequency of challenge. 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. As stated in NEI 00-04, an SSC identified as HSS by a non-PRA method to address fire “may not be re-categorized by the IDP.”

Furthermore, regulatory exemptions related to the Fire Safe Shutdown program and previously identified fire-induced MSOs were reviewed and additional equipment that is relied upon to establish and maintain safe shutdown will be retained as HSS. The results of this review have been documented by the site and are available for NRC audit. Figure 3-1 illustrates how the Fire SSEL and the additional identified equipment are reviewed to determine if the component being evaluated is HSS.

This approach is an alternate process from the NEI 00-04 endorsed approaches. Similar to the NEI 00-04 FIVE approach, this approach uses the SSEL as a screening tool. However, the development of the Fire SSEL is not based on a successive screening methodology and is the starting point for the FIVE methodology. Therefore, industry assessments have shown that this Fire SSEL approach leads to many additional SSCs being identified as HSS making it more conservative in determining safety significance than the NEI 00-04 FIVE approach or a Fire PRA.

The SQN Fire Safe Shutdown program is an active regulatory program that is routinely inspected by NRC. It was confirmed that this program ensures that the Fire SSEL and the identification of additional equipment relied upon to establish and maintain safe shutdown reflects the current as-built, as-operated plant and that changes to the plant will be evaluated to determine their impact to the equipment list and the categorization process.

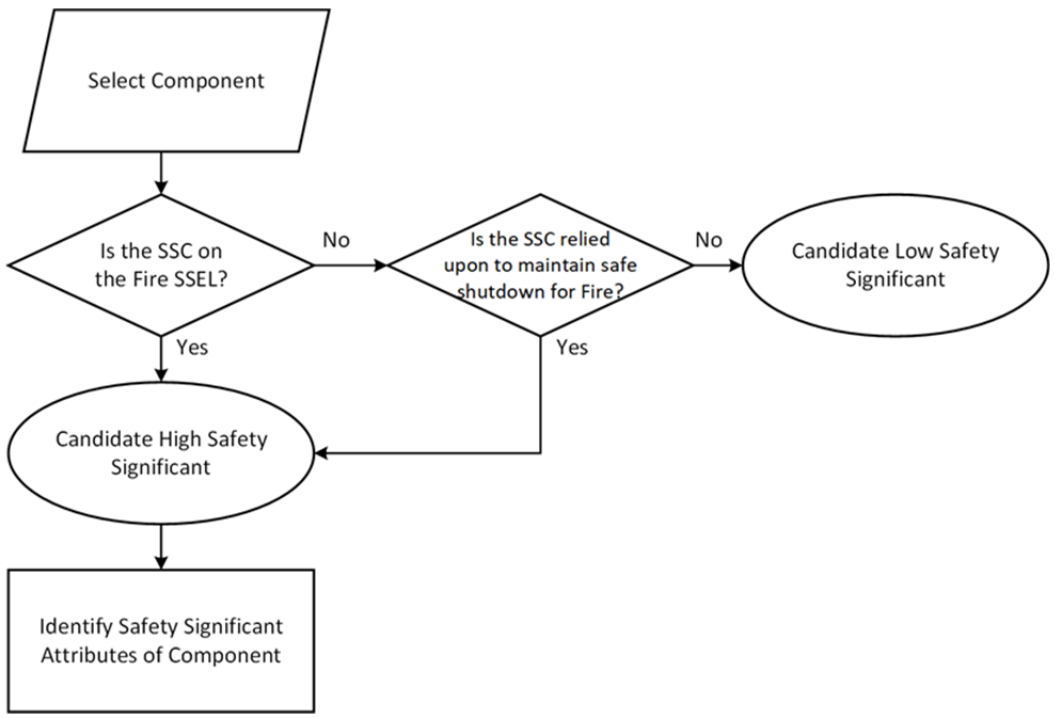

Figure 3.1: Safety Significance Process for Structures, Systems, and Components Addressed in Fire Safe Shutdown Program

3.2.3 Seismic Hazards

The SQN categorization process will use the SMA performed for the IPEEE in response to Generic Letter (GL) 88-20 Supplement 4 (Reference12) for evaluation of safety significance related to seismic hazards. No plant specific approaches were utilized in the development of the SMA. The NEI 00-04 approved use of the SMA SSEL as a screening process results in the categorization of all system functions and associated SSCs that are involved in the seismic margin success path as HSS. Because the

analysis is being used as a screening tool, importance measures are not used to determine safety significance. The NEI 00-04 approach using the SMA SSEL identifies credited equipment as HSS regardless of their capacity, frequency of challenge, or level of functional diversity.

An evaluation was performed of the as-built, as-operated plant against the SMA SSEL. The evaluation compared the as-built, as-operated plant to the plant configuration originally assessed by the SMA. Differences were reviewed to identify any potential impacts to the equipment credited on the SMA SSEL. Appropriate changes to the credited equipment were identified and documented. This documentation is available for audit. The TVA risk management program ensures that future changes to the plant will be evaluated to determine their impact on the SMA and risk categorization process.

3.2.4 Other External Hazards

The SQN categorization process will use the screening results from the IPEEE in response to GL 88-20 (Reference 12) for the evaluation of safety significance related to the following other external hazards:

- High Winds
- External Flooding
- Transportation and Nearby Facility Accidents
- Other External Initiating Events

Figure 5-6 in Section 5.4 of NEI 00-04 illustrates the process that will be used to determine safety significance related to the above hazards.

All other external hazards were screened from applicability to SQN Units 1 and 2 per a plant-specific evaluation in accordance with GL 88-20 and updated to use the criteria in ASME PRA Standard RA-Sa-2009 (Reference 13). Attachment 4 provides a summary of the other external hazards screening results. Attachment 5 provides a summary of the progressive screening approach for external hazards.

As part of the categorization assessment of other external hazard risk, an evaluation is performed to determine if there are components being categorized that participate in screened scenarios and whose failure would result in an unscreened scenario. Consistent with the flow chart in Figure 5-6 in Section 5.4 of NEI 00-04, these components would be considered HSS.

3.2.5 Low Power & Shutdown

Consistent with NEI 00-04, the SQN categorization process will use the shutdown safety management plan described in NUMARC 91-06 (Reference 8), for evaluation of safety significance related to low power and shutdown conditions.

The overall process for addressing shutdown risk is illustrated in Figure 5-7 of NEI 00-04.

NUMARC 91-06 specifies that a defense-in-depth approach should be used with respect to each defined shutdown key safety function. The key safety functions defined in NUMARC 91-06 are evaluated for categorization of SSCs.

SSCs that meet the two criteria (i.e., considered part of a “primary shutdown safety system” or a failure would initiate an event during shutdown conditions) described in Section 5.5 NEI 00-04 will be considered preliminary HSS.

3.2.6 PRA Maintenance and Updates

The TVA risk management process ensures that the applicable PRA model(s) used in this application continues 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, and 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 will implement a process that addresses the requirements in NEI 00-04, Section 11, “Program Documentation and Change Control.” The process will review 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 will be peer reviewed prior to implementing those changes in the PRA model used for categorization.

3.2.7 PRA Uncertainty Evaluations

Uncertainty evaluations associated with any applicable baseline PRA model(s) used in this application were evaluated during the assessment of PRA technical adequacy and confirmed through the self-assessment and peer review processes as discussed in Section 3.3 of this enclosure.

Uncertainty evaluations associated with the risk categorization process are addressed using the processes discussed in NEI 00-04 Section 8 “Risk Sensitivity Study,” and in the prescribed sensitivity studies discussed in Section 5 “Component Safety Significance Assessment.”

In the overall risk sensitivity studies, TVA will utilize a factor of three to increase the unavailability or unreliability of LSS components consistent with that approved for the Vogtle Electric Generating Plant (Reference 10). Consistent with the NEI 00-04 guidance, TVA will perform 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 all identified PRA models for all systems that have been categorized are increased by a factor of three. 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.

The detailed process of identifying, characterizing, and qualitative screening of model uncertainties is found in Section 5.3 of NUREG-1855 (Reference 14) and Section 3.1.1 of EPRI TR-1016737 (Reference 15). The process in these references was mostly developed to evaluate the uncertainties associated with the internal events PRA model; however, the approach can be applied to other types of hazard groups.

The list of assumptions and sources of uncertainty were reviewed to identify those which would be significant for the evaluation of this application. If the SQN PRA model used a non-conservative treatment, or methods that are not commonly accepted, the underlying assumption or source of uncertainty was reviewed to determine its impact on this application. Only those assumptions or sources of uncertainty that could significantly impact the configuration risk calculations were considered key for this application.

Key SQN PRA model specific assumptions and sources of uncertainty for this application were identified and dispositioned in Attachment 6. This information is available for NRC audit. The conclusion of this review is that no additional sensitivity analyses are required to address SQN PRA model specific assumptions or sources of uncertainty.

3.3 PRA REVIEW PROCESS RESULTS (10 CFR 50.69(b)(2)(iii))

The PRA models described in Section 3.2.1 has been assessed against RG 1.200, "An Approach for Determining the Technical Adequacy of Probabilistic Risk Assessment Results for Risk-Informed Activities," Revision 2 (Reference 16) consistent with NRC Regulatory Issue Summary (RIS) 2007-06 (Reference 17).

The internal events PRA model was subjected to a self-assessment and a full-scope peer review conducted in January 2011.

A finding closure review was conducted on the identified PRA models on May 8 to May 10, 2017. Closed findings were reviewed and closed using the process documented in the NEI letter to the NRC "Final Revision of Appendix X to NEI 05-04/07-12/12-16, Close-Out of Facts and Observations (F&O)," (Reference 18), as accepted by NRC in Reference 19. The results of this review have been documented and are available for NRC audit. SQN has no open Findings (see Attachment 3). Please note that SQN has no open F&O findings.

The information above demonstrates that the PRA is of sufficient quality and level of detail to support the categorization process, and has been subjected to a peer review process assessed against a standard or set of acceptance criteria that is endorsed by the NRC as required by 10 CFR 50.69(c)(1)(i).

3.4 RISK EVALUATIONS (10 CFR 50.69(b)(2)(iv))

The SQN 10 CFR 50.69 categorization process will implement the guidance in NEI 00-04. 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 NEI 00-04 Section 8 will be used to confirm that the categorization process results in acceptably small increases to core damage frequency (CDF) and large early release frequency (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, and human errors). 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.

4. REGULATORY EVALUATION

4.1 APPLICABLE REGULATORY REQUIREMENTS/CRITERIA

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

- The regulations in 10 CFR Part 50.69, "Risk-Informed Categorization and Treatment of Structures, Systems and Components for Nuclear Power Reactors" (Reference 20)
- RG 1.201, "Guidelines for Categorizing Structures, Systems, and Components in Nuclear Power Plants According to their Safety Significance," Revision 1, May 2006 (Reference 6).
- RG 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 (Reference 21).
- RG 1.200, "An Approach for Determining the Technical Adequacy of Probabilistic Risk Assessment Results for Risk-Informed Activities," Revision 2, March 2009" (Reference 16).

The proposed change is consistent with the applicable regulations and regulatory guidance.

4.2 NO SIGNIFICANT HAZARDS CONSIDERATION ANALYSIS

Tennessee Valley Authority (TVA) proposes to modify the licensing basis to allow for the 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 for Nuclear Power Reactors." 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 not be

changed or will be enhanced. This allows improved focus on equipment that has safety significance resulting in improved plant safety.

TVA has evaluated whether or not a significant hazards consideration is involved with the proposed amendment(s) by focusing on the three standards set forth in 10 CFR 50.92, "Issuance of amendment," as discussed 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 will permit the use of a risk-informed categorization process to modify the scope of structures, systems, and components (SSCs) subject to Nuclear Regulatory Commission (NRC) special treatment requirements and to implement alternative treatments per the regulations. 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 significantly 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 will 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 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 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.

4.3 CONCLUSIONS

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 the Commission's 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. 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 a 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. REFERENCES

1. Nuclear Energy Institute, NEI 00-04, "10 CFR 50.69 SSC Categorization Guideline," Revision 0, dated July 2005
2. TVA letter to NRC, "Sequoyah Nuclear Plants Units 1 and 2 Technical Specifications Conversion to NUREG 1431, Revision 4.0 (SQN-TS-11-10)," dated November 22, 2013 (ML13329A881)
3. NRC Letter to TVA, "Sequoyah Nuclear Plant, Units 1 and 2 - Issuance of Amendments for the Conversion to the Improved Technical Specifications with Beyond Scope Issues (TAC Nos. MF3128 and MF3129)," dated September 30, 2015 (ML15238B460)
4. Technical Specification Task Force (TSTF) - 425, Revision 3, "Relocate Surveillance Frequencies to Licensee Control - RITSTF Initiative 5b," dated March 18, 2009 (ML090850627)
5. TSTF-446, "Risk Informed Evaluation of Extensions to Containment Isolation Valve Completion Times (WCAP-15791)," Revision 3, dated February 19, 2008 (ML080510164)
6. Regulatory Guide 1.201, "Guidelines for Categorizing Structures, Systems, and Components in Nuclear Power Plants According to their Safety Significance," Revision 1, dated May 2006 (ML061090627)
7. NRC Letter to TVA, "Sequoyah Nuclear Plant, Units 1 and 2 - Review of Sequoyah Individual Plant Examination of External Events Submittal (TAC Nos. M83674 and M83675)," dated February 21, 2001 (ML010520133)
8. NUMARC 91-06, "Guidelines for Industry Actions to Assess Shutdown Management," dated December 1991 (ML14365A203)
9. NRC Letter to Entergy, "Arkansas Nuclear One, Unit 2 - Approval of Request for Alternative ANO2-R&R-004, Revision 1, Request to Use Risk-Informed Safety Classification and Treatment for Repair/Replacement Activities in Class 2 and 3 Moderate and High /Energy Systems (TAC No. MD5250)," dated April 22, 2009 (ML090930246)
10. NRC Letter to Southern Nuclear Operating Company, Inc., "Vogtle Electric Generating Plant, Units 1 and 2 - Issuance of Amendments Re: Use of 10 CFR 50.69 (TAC Nos. ME9472 and ME9473)," dated December 17, 2014 (ML14237A034)
11. Branch Technical Position CMEB 9.5-1, "Guidelines for Fire Protection for Nuclear Power Plants," Revision 3, dated July 1981 (ML070660454)
12. Generic Letter 88-20, "Individual Plant Examination of External Events (IPEEE) for Severe Accident Vulnerabilities - 10 CFR 50.54(f), Supplement 4," dated June 1991
13. ASME/ANS RA-Sa-2009, "Addenda to ASME/ANS RA-S-2008, Standard for Level 1/Large Early Release Frequency Probabilistic Risk Assessment for Nuclear Power Plant Applications," dated February 2, 2009

14. NUREG-1855, “Guidance on the Treatment of Uncertainties Associated with PRAs in Risk-Informed Decision Making,” dated March 2009 (ML090970525)

15. EPRI TR-1016737, “Treatment of Parameter and Model Uncertainty for Probabilistic Risk Assessments,” dated December 2008

16. Regulatory Guide 1.200, “An Approach for Determining the Technical Adequacy of Probabilistic Risk Assessment Results for Risk-Informed Activities,” Revision 2, March 2009 (ML090410014)

17. NRC Regulatory Issue Summary 2007-06, “Regulatory Guide 1.200 Implementation,” dated March 22, 2007 (ML070650428)

18. 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),” dated February 21, 2017 (ML17086A431)

19. NRC Letter to Mr. Greg Krueger (NEI), “U.S. Nuclear Regulatory Commission Acceptance on Nuclear Energy Institute Appendix X to Guidance 05-04, 7-12, and 12-13, Close-Out of Facts and Observations (F&Os),” dated May 3, 2017 (ML17079A427)

20. Title 10 of the Code of Federal Regulations (10 CFR) Part 50.69, "Risk-Informed Categorization and Treatment of Structures, Systems and Components for Nuclear Power Reactors”

21. 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, dated April 2015.

Attachment 1: List of Categorization Prerequisites

TVA will establish procedure(s) prior to the use of the categorization process on a plant system. The procedure(s) will contain the elements/steps listed below.

- Independent Decision-Making Panel (IDP) member qualification requirements.

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

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

- Assessment of defense-in-depth (DID) and safety margin. Components that are categorized·as preliminary LSS, are evaluated for their role in providing DID and safety margin and, if appropriate, upgraded to HSS.

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

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

- Internal Event Risks: Internal Events including internal flooding PRA model Revision 3, dated August 5, 2014 or later updated model as described in section 3.2.6 of this enclosure will be used. This model was accepted by NRC in Reference 3.

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

- Documentation requirements per Section 3.1.1 of this enclosure.

Attachment 2: Description of PRA Model Used in Categorization

Units	Model	Baseline CDF	Baseline LERF	Comments
1 & 2	SQN Units 1 & 2 Integrated Model for Internal Events and Internal Flooding Hazards - CAFTA Rev. 3, August 2014	Unit 1 1.56E-05/yr Unit 2 1.63E-05/yr	Unit 1 2.61E-06/yr Unit 2 2.66E-06	CAFTA Rev. 0 Peer Reviewed Against RG 1.200 Rev. 2, Jan 2011. The following model updates have been performed subsequent to the 2011 Peer Review: Rev. 1 Aug 2012 Rev. 2 Dec 2013 Rev. 3 Aug 2014 The technical adequacy of the CAFTA Model Rev. 3 was previously approved by NRC in the ITS SER (Reference 3).

Attachment 3: Disposition and Resolution of Open Peer Review Findings and Self-Assessment Open Items

SQN has no open Facts and Observation (F&O) Findings. The closure review was performed during the week of May 8, 2017.

Attachment 4: External Hazards Screening			
External Hazard	Screening Result		
	Screened? (Y/N)	Screening Criterion (Note a)	Comment
Aircraft Impact	Y	PS2 PS4	The Dallas Bay Sky Park is located beyond five miles of the plant. Also, Federal Airway V333 passes directly over the site. The Chattanooga Airport is located approximately 14.5 miles from the plant site. These are the only facilities of potential significance to the safe operation of the plant, and based on evaluations these activities will pose no hazard.
Avalanche	Y	C3	SQN is located in the Tennessee Valley on the Tennessee River. Avalanches are not a viable external initiator because of climate and topography, as SQN is located within a temperate zone.
Biological Event	Y	C5	Sudden influxes are not applicable to the plant design. Slowly developing growth can be detected and mitigated by surveillance. Control of organic fouling is provided by use of strainers in the supply headers and biocide treatments. Asiatic clams are controlled by a combination of straining and biocide treatments. Microbiologically induced corrosion (MIC) is controlled by injecting biocide.
Coastal Erosion	Y	C3	SQN is located in the Tennessee Valley; therefore, coastal erosion is not a concern as a credible hazard.

Attachment 4: External Hazards Screening			
External Hazard	**Screening Result**		
	Screened? (Y/N)	**Screening Criterion (Note a)**	**Comment**
Drought	Y	C5	Plant design eliminates drought as a concern. In addition, this event is slowly developing. TVA's management of the Tennessee River level is such that drought will not jeopardize the ultimate heat sink's capability.
External Flooding	Y	PS1 PS4	Flood protection plans, designed to minimize impact of floods above plant grade on safety-related facilities are in place, and procedures for predicting rainfall floods, arrangements to warn of upstream dam failure floods, and lead times available and types of action to be taken to meet safety requirements for sources are thoroughly analyzed and compensatory measures are well planned. Some wind driven waves are likely when the probable maximum flood (PMF) crests at SQN. Analysis shows that the probability that this would occur on the specific day of the PMF is on the order of 1E-10 or lower. Considering the simultaneous events over a 40-yr period is on the order of 1E-06. Therefore, wind driven waves are not a concern to plant safety. Snowmelt is not a factor in generating maximum floods at the plant site due to the plant location, which is within a temperate zone. SQN conforms to regulatory position of RG 1.59, "Design Basis Floods for Nuclear Power Plants."

Attachment 4: External Hazards Screening			
External Hazard	Screening Result		
	Screened? (Y/N)	Screening Criterion (Note a)	Comment
Extreme Wind or Tornado	Y	PS1 PS3	The SQN plant Category I structures are designed for a 95-mph wind (including a 1.1 gust factor) 30 feet above grade with a 100-year period of recurrence. The basis was determined from American Society of Civil Engineers (ASCE) Paper 3269, "Wind Forces on Structures." The SQN plant is designed for tornados having a maximum rotational velocity of 300 mph, and up to a translational speed of 60 mph. SQN has been designed to resist tornado wind and missile effects equivalent to the Design Basis Tornado and meets the intent of RG 1.76 and 1.117.
Fog	Y	C4	Fog and mist may increase the frequency of accidents involving aircraft, ships, or vehicles. This weather condition is included implicitly in the accident rate for these Transportation Accidents
Forest or Range Fire	Y	C3	The ground has been cleared for two-thousand feet around plant buildings. There are no wooded areas close enough to present a hazard from forest fires.
Frost	Y	C4	Implicitly included in weather-related LOOP, such as hail and snow.

Attachment 4: External Hazards Screening			
External Hazard	**Screening Result**		
	Screened? (Y/N)	**Screening Criterion (Note a)**	**Comment**
Hail	Y	C1 C4	Hail events could cause complete or partial losses of offsite power (LOOP) (Reference: NUREG/CR-5042 Supplement 2). The potential effects of a LOOP and station blackout are addressed by the Internal Events PRA and the frequency of hail events is subsumed by the LOOP initiating event frequency for weather related events. Therefore, this event is bounded by other events for which the plant is designed.
High Summer Temperature	Y	C1 C4	SQN meets General Design Criteria 22 and 23 that require the design of protection systems assures that the effects of natural phenomena and other conditions do not result in a loss of the protection function. Furthermore, the design of protection systems fail in the safe state or into a state determined to be acceptable on some other defined basis if conditions such as disconnection of the system, loss of energy (e.g., electric power, instrument air) or postulated adverse environments (e.g., extreme heat, cold) are experienced.
High Tide, Lake Level, or River Stage	Y	C4	SQN conforms to regulatory position of RG 1.59, “Design Basis Floods for Nuclear Power Plants.” See External Flooding for additional information.

Attachment 4: External Hazards Screening			
External Hazard	Screening Result		
	Screened? (Y/N)	Screening Criterion (Note a)	Comment
Hurricane	Y	C3 C4	Due to the location of the SQN plant in the Tennessee Valley, a hurricane is not a credible event. However, the downgraded storm is expected to cause other hazards that are covered under Extreme Wind or Tornado, and Intense Precipitation (i.e. External Flooding).
Ice Cover	Y	C3	Because of SQN's location in a temperate climate, significant amounts of ice do not form on Tennessee Valley rivers and lakes.
Industrial or Military Facility Accident	Y	C3	There are no industrial or military facilities within five miles of the SQN plant that would potentially pose a hazard to the safe operation of the plant. Facilities of interest beyond five miles include the Volunteer Army Ammunition plant that is the only industrial or military facility of potential significance to the safe operation of the plant.
Internal Flooding	N	N/A	SQN has an Internal Flooding Model
Internal Fire	N	N/A	Addressed in Section 3.2.2
Landslide	Y	C3	Not applicable to SQN because of topography.

Attachment 4: External Hazards Screening			
External Hazard	**Screening Result**		
	Screened? (Y/N)	**Screening Criterion (Note a)**	**Comment**
Lightning	Y	C1 C4	Lightning strikes causing loss of offsite power or turbine trip are contributors to the initiating event frequencies for these events. However, other causes are also included in the loss of offsite power initiator. Impacts from lightning are no greater than already modeled in the internal events PRA.
Low Lake Level or River Stage	Y	C5	This event is slowly developing and well controlled based on TVA's management of the Tennessee River.
Low Winter Temperature	Y	C1 C5	Extended freezing temperatures are rare in the Tennessee Valley where the SQN plant is located. The plant is designed for such events, and their impacts are slow to develop.
Meteorite or Satellite Impact	Y	PS4	Extraterrestrial activity includes both natural and manmade objects that enter earth's atmosphere from space. Because the probability of a meteorite strike is very small. Therefore, it can be dismissed on the basis of its very low initiating event frequency.
Pipeline Accident	Y	C3	There are no major natural gas pipelines within five miles of SQN. The nearest eight-inch diameter pipeline is located at more than 0.5 miles at its closest approach to the site. Any postulated rupture of this pipe at that distance would not pose a hazard to the safety-related SSCs at SQN.

Attachment 4: External Hazards Screening			
External Hazard	Screening Result		
	Screened? (Y/N)	Screening Criterion (Note a)	Comment
Release of Chemicals in Onsite Storage	Y	C2	The main control room habitability during postulated hazardous chemical releases at or near the plant was evaluated by TVA. The evaluation conformed to RG 1.78 and concluded that the main control room habitability is not jeopardized by accident releases of the chemicals. SQN maintains a list of hazardous materials, storage location, and quantities. There are no industrial or military facilities where large quantities of toxic chemicals could be stored with a 5-mile radius of the plant.
River Diversion	Y	C3	River (channel) diversion is not a potential problem for SQN. There are no channel diversions upstream of the plant that would cause diverting or rerouting of the source of plant cooling water, and none are anticipated in the future. The flood plain is such that large floods do not produce major channel meanders or cutoffs. Analyses indicate the Tennessee River has essentially maintained its present alignment for over 35,000 years.
Sand or Dust Storm	Y	C3	Due to plant location, SQN is not subject to sand or dust storms.
Seiche	Y	C3	SQN is located on the Tennessee River (Chickamauga Lake). No conceivable hurricane or cyclonic-type winds could produce wave heights required to reach the plant grade.

Attachment 4: External Hazards Screening			
External Hazard	Screening Result		
	Screened? (Y/N)	Screening Criterion (Note a)	Comment
Seismic Activity	N	N/A	NRC's IPEEE Safety Evaluation for SQN dated February 21, 2001 (Reference 7) acknowledged that SQN was categorized as a full-scope plant (per NUREG-1407), having a 0.3g peak ground acceleration. SQN performed a full-scope seismic margin analysis using the methodology described in EPRI NP-6041-SL, Revision 1, "A Methodology for Assessment of Nuclear Power Plant Seismic Margin." The NRC Staff concluded that the aspect of seismic hazards was adequately addressed for SQN.
Snow	Y	C3	Roofs of structures at SQN are designed for snow loading greater than observed or expected for the location of the plant site, which is within a temperate zone. Potential flooding impacts covered under external flooding.
Soil Shrink-Swell Consolidation	Y	C1 C5	The SQN Updated FSAR, Chapter 2.5 describes the characteristics of the stratigraphy, rocks, foundation, soils, and backfill. For Category I soil-supported structures, the allowable capacity for sustained loading is at least a factor of three less than the ultimate bearing. The potential for this hazard is low at the SQN site.
Storm Surge	Y	C3	SQN is located in the Tennessee Valley on the Tennessee River; therefore storm surge is not a viable external initiator for SQN.

Attachment 4: External Hazards Screening			
External Hazard	**Screening Result**		
	Screened? (Y/N)	**Screening Criterion (Note a)**	**Comment**
Toxic Gas	Y	C2	The main control room habitability during postulated hazardous chemical releases at or near the plant was evaluated by TVA. The evaluation conformed to RG 1.78 and concluded that the main control room habitability is not jeopardized by accident releases of the chemicals. SQN maintains a list of hazardous materials, storage location, and quantities. There are no industrial or military facilities where large quantities of toxic chemicals could be stored with a 5-mile radius of the plant.

Attachment 4: External Hazards Screening			
External Hazard	Screening Result		
	Screened? (Y/N)	Screening Criterion (Note a)	Comment
Transportation Accident	Y	 PS2 C4 C3 PS4	An examination of the impact from potential accidents on the transportation routes (i.e., railroad, aircraft, highways, and barge traffic) concluded that their contribution to plant risk is negligible. Railroads -As noted in SER for the SQN IPEEE Section 2.3.3.2, (Reference 7) “The nearest mainline railroad is 5.5 miles away, which is greater than the Regulatory Guide 1.91 safe distance.” There is a spur that leads into the plant; however, it is limited to delivery of large components to the plant on an infrequent basis. Aircraft - see Aircraft Impact Highways - As noted in Reference 7, Section 2.3.3.2, “The nearest highway is two miles away, which is greater than the RG 1.91 safe distance.” Barge Traffic - The potential for damage to the SQN plant from a barge explosion is negligible. Materials considered included TNT, gasoline, liquid natural gas and fertilizers. Barge impacts on plant structures are conservatively calculated to be in the 1E-05/yr (upstream tow), and 1E-08/yr (Downstream Drifting) range.

Attachment 4: External Hazards Screening			
External Hazard	**Screening Result**		
	Screened? (Y/N)	**Screening Criterion (Note a)**	**Comment**
Tsunami	Y	C3	SQN is located in the Tennessee Valley, which is not adjacent to any large bodies of water. Therefore, a tsunami event is not applicable to SQN because of location.
Turbine-Generated Missiles	Y	PS2	The SQN plant arrangement is such that safety-related SSCs are essentially protected from low trajectory turbine missiles. The probability of this event has been estimated at less than 1E-07/yr. SQN has an inspection program of the low pressure turbine discs that is performed on a regular basis.
Volcanic Activity	Y	C3	SQN is located in the Tennessee Valley which is not prone to volcanic activity.
Waves	Y	C3 C4	SQN is located in the Tennessee Valley and not adjacent to any large body of water that can result in significant waves. Therefore large wave events are not applicable to the SQN site. Note: Waves associated with External Flooding are covered under that hazard.
Note a – See Attachment 5 for descriptions of the screening criteria.			

Attachment 5: Progressive Screening Approach for Addressing External Hazards			
Event Analysis	**Criterion**	**Source**	**Comments**
Initial Preliminary Screening	C1. Event damage potential is less than events for which plant is designed.	NUREG/CR-2300 and ASME/ANS Standard RA-Sa-2009	
	C2. Event has lower mean frequency and no worse consequences than other events analyzed.	NUREG/CR-2300 and ASME/ANS Standard RA-Sa-2009	
	C3. Event cannot occur close enough to the plant to affect it.	NUREG/CR-2300 and ASME/ANS Standard RA-Sa-2009	
	C4. Event is included in the definition of another event.	NUREG/CR-2300 and ASME/ANS Standard RA-Sa-2009	Not used to screen. Used only to include within another event.
	C5. Event develops slowly, allowing adequate time to eliminate or mitigate the threat.	ASME/ANS Standard RA-Sa-2009	
Progressive Screening	PS1. Design basis hazard cannot cause a core damage accident.	ASME/ANS Standard RA-Sa-2009	
	PS2. Design basis for the event meets the criteria in the NRC 1975 Standard Review Plan (SRP).	NUREG-1407 and ASME/ANS Standard RA-Sa-2009	

Attachment 5: Progressive Screening Approach for Addressing External Hazards			
Event Analysis	**Criterion**	**Source**	**Comments**
	PS3. Design basis event mean frequency is < 1E-05/yr. and the mean conditional core damage probability is < 0.1.	NUREG-1407 as modified in ASME/ANS Standard RA-Sa-2009	
	PS4. Bounding mean CDF is <1E-06/yr.	NUREG-1407 and ASME/ANS Standard RA-Sa-2009	
Detailed PRA	Screening not successful. PRA needs to meet requirements in the ASME/ANS PRA Standard.	NUREG-1407 and ASME/ANS Standard RA-Sa-2009	

Attachment 6: Disposition of Key Assumptions/Sources of Uncertainty		
Assumption/Uncertainty	**Discussion**	**Disposition**
Internal Flooding - The assumption that a leaking pipe will be detected by visual inspection is given a 0.9 probability. Similarly, detection probabilities for Non-Destructive Testing (NDT) uses probabilities up to 0.9.	The assumed 0.9 value is consistent with EPRI Report 3002000079 R3, Pipe Rupture Frequencies for Internal Flooding PRA. Areas with potentially HSS SSCs could be sensitive to higher flooding probabilities. Since internal flooding is a significant contributor to plant risk, and the detection probability is significant to reducing the impact of a flooding scenario, detection is a major factor in the analysis.	Uncertainty evaluations associated with the risk categorization process are addressed using the processes discussed in NEI 00-04 Section 8 and in the prescribed sensitivity studies discussed in Section 5. The TVA process will follow those requirements to address these and similar assumptions.
The Internal Flooding analysis calculation uses a summation of three different frequencies, passive pipe break failures, human-induced flooding, and maintenance induced flooding.	Each of these flooding events has its own inherent uncertainties.	Passive pipe break failures rates have been given an uncertainty parameter. The impact of these uncertainties can be treated by the use of a random sampling Monte Carlo process. Human induced flooding is analyzed by use of the HRA Calculator program which creates an assumed uncertainty term for any HRA action. Since the human induced flooding events is a combination of both pre-initiating event and post initiating event, each portion has an independent uncertainty term. The HRA Calculator program also arbitrarily assigns an uncertainty term to HRA actions based on the calculated probabilities. Maintenance induced flooding events has three main inputs to the calculation of this frequency, failure rate of an MOV, mission time, and frequency of the activity, of

Attachment 6: Disposition of Key Assumptions/Sources of Uncertainty		
Assumption/Uncertainty	**Discussion**	**Disposition**
		which, all introduce some level of uncertainty into the calculation. The large internal rupture of an MOV is assumed in NUREG/CR-6928 to be a factor of 0.02 less than that of a small internal leak on an MOV as there has been no actual large internal rupture events in the industry. The mission time is also assumed based on a seven day repair interval, this number could potentially be greater than that if the component is not covered by an Technical Specification or, more likely, less than the assumed seven day repair time. The final area of uncertainty is the frequency of the activity. Most of the procedures reviewed have frequencies as well as conditions. These conditions could cause the actual maintenance activity to occur more often than the frequency noted in the procedure.
Equipment type code data includes successful post-maintenance testing (PMT).	Inclusion of successful PMT demands can result in an under-estimation of the failure probability of a component type.	This was addressed by performance of sensitivity studies that indicate inclusion of the number of assumed PMT demands has a small impact on CDF. If the number of successful PMT starts is significantly overestimated it can have a more profound affect on CDF and LERF.
State of Knowledge Correlation (SOKC).	The SOKC causes point estimates to differ from the mean values, which are higher. SOKC is a statistical effect that appears when a pool of data is used to characterize the uncertainty distribution for all events within the data pool. Those events that are used are considered correlated, which implies that the	To account for the potential correlation of various type codes, factors are employed that multiply the value of the cutset to ensure the resultant frequency is in line with the mean. These factors were calculated using the multiplier method, where an additional basic event is appended to the cutsets with correlated basic events. Since the normal cutset frequency calculation is

Attachment 6: Disposition of Key Assumptions/Sources of Uncertainty		
Assumption/Uncertainty	**Discussion**	**Disposition**
	same distribution applies to all sampled events when using a Monte Carlo approach.	performed via the multiplication method, this eases use of these factors. The factors were based on use of the multiplier method discussed in WCAP 17154-P. This method requires the recovery rules to be written to add factors onto each applicable cutset. The factors are calculated based on the uncertainty distribution type and the associated alpha and beta shaping factors. Both the Beta distribution and the Gamma distribution were calculated for the SOKC.

Enclosure 2

Proposed Changes (Mark-Ups) to SQN Renewed Operating Licenses for Units 1 and Unit 2

(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.

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

(33) 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 (SSC) specified in the license amendment dated [XXXX].

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).

TVA shall complete the items listed in Attachment 1, List of Categorization Prerequisites, of TVA letter dated [XXXX], prior to implementation.

... fect all provisions of the ... d qualification, and ... s made pursuant to ... earch Requirements ...) and to the authority of 10 ... t of plans, which contain ... 21, is entitled: "Sequoyah ... on Plan, And Safeguards ... 2006.

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

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

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

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

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

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

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

(26) TVA 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 (SSC) specified in the license amendment dated [XXXX].

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).

TVA shall complete the items listed in Attachment 1, List of Categorization Prerequisites, of TVA letter dated [XXXX], prior to implementation.

Enclosure 3

Proposed Changes (Final Typed) to SQN Renewed Operating Licenses for Unit 1 and Unit 2

(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) 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 (SSC) specified in the license amendment dated [XXXX].

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).

TVA shall complete the items listed in Attachment 1, List of Categorization Prerequisites, of TVA letter dated [XXXX], prior to implementation.

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

E. Physical Protection

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

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

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

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

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

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

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

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

(26) TVA 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 (SSC) specified in the license amendment dated [XXXX].

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).

TVA shall complete the items listed in Attachment 1, List of Categorization Prerequisites, of TVA letter dated [XXXX], prior to implementation.