Southern Nuclear

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10 CFR 50.90
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U. S. Nuclear Regulatory Commission
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Joseph M. Farley 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"

In accordance with the provisions of 10 CFR 50.69 and 10 CFR 50.90, Southern Nuclear Operating Company (SNC) is requesting an amendment to the license of Farley Nuclear Plant (FNP).

The proposed amendment would modify the FNP licensing basis, by the addition of a License Condition, 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.

The enclosure to this letter provides the basis for the proposed change to the FNP 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 will only occur after these prerequisites are met.

SNC requests approval of the proposed license amendment within one year of the date of this letter, with the amendment being implemented within 90 days following NRC approval.

In accordance with 10 CFR 50.91, a copy of this application, with attachments, is being provided to the designated Alabama State Official.

This letter contains no regulatory commitments.

Should you have any questions concerning this submittal, please contact Jamie Coleman at 205.992.6611.

I declare under penalty of perjury that the foregoing is true and correct. Executed on the 18th day of June 2020.

Respectfully submitted,

Cheryl Gayheart
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CAG/RMJ

Enclosure: Evaluation of the Proposed Change

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Joseph M. Farley 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"

Enclosure

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LIST OF ATTACHMENTS

1 SUMMARY DESCRIPTION

The proposed amendment modifies the 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 (LSS), alternative treatment requirements can be implemented in accordance with this regulation. For equipment determined to be of high safety significance (HSS), 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," used principally in the general design criteria (GDC) of Appendix A to 10 CFR Part 50, is not explicitly defined.

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, 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 SNC to improve focus on equipment that has safety significance resulting in improved plant safety.

2.3 DESCRIPTION OF THE PROPOSED CHANGE

SNC proposes the addition of the following condition to the renewed operating license of FNP to document the NRC's approval of the use 10 CFR 50.69.

> SNC 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) models to evaluate risk associated with internal events, including internal flooding, and internal fire; the 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 and non-Class SSCs and their associated supports; the results of the non-PRA evaluations that are based on the IPEEE Screening Assessment for External Hazards updated using the external hazard screening significance process identified in ASME/ANS PRA Standard RA-Sa-2009 for other external hazards except seismic; and the alternative seismic approach as described in SNC's submittal letter dated June 18, 2020, and all its subsequent associated supplements as specified in License Amendment No. [XXX] dated [DATE].
>
> 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).

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.

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

3.1.1 Overall Categorization Process

SNC will implement the risk categorization process in accordance with 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 [2]). 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.

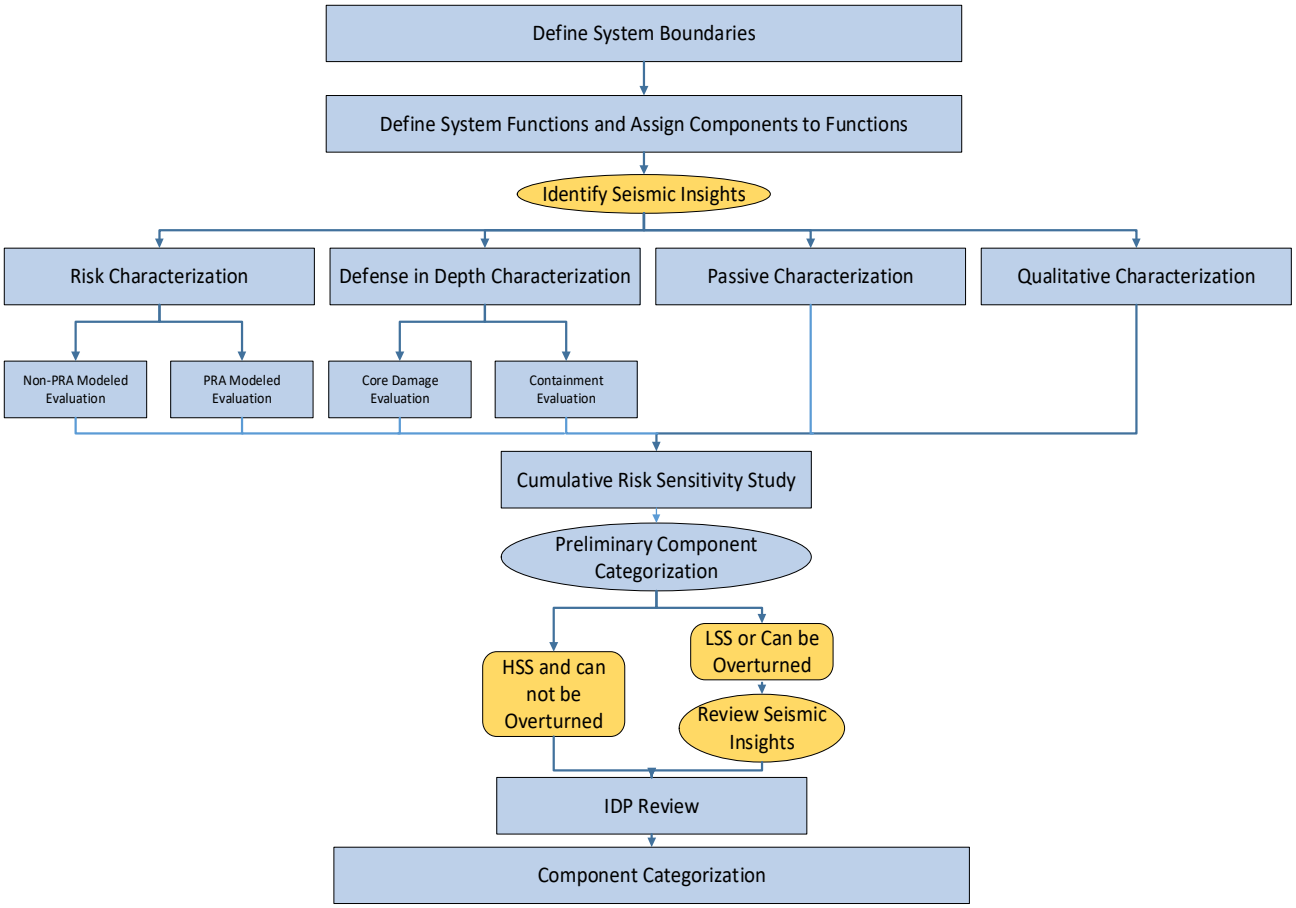
The process to categorize each system will be consistent with the guidance in NEI 00-04, "10 CFR 50.69 SSC Categorization Guideline," as endorsed by RG 1.201, with the exception of the evaluation of impact of the seismic hazard, which will use the EPRI 3002017583 (Reference [3]) approach for seismic Tier 1 sites, which includes FNP, to assess seismic hazard risk for 50.69. Inclusion of additional process steps discussed below to address

seismic considerations will ensure that reasonable confidence in the evaluations required by 10 CFR 50.69(c)(1)(iv) is achieved. 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 complete they may even be performed in parallel. Note that NEI 00-04 only requires Item 3 to be completed for components/functions categorized as Low Safety Significant (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, Seismic Safe Shutdown Equipment List, 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

Figure 3-1 is an example of the major steps of the categorization process described in NEI 00-04; two steps (represented by four blocks on the figure) have been included to highlight review of seismic insights as pertains to this application, as explained further in Section 3.2.3:

Figure 3-1: Categorization Process Overview



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., HSS or 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 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: Categorization Evaluation Summary

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 and Other External Hazards –	Component	Not Allowed	No
	Seismic –	Function/Component	Allowed [2]	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[1]	N/A
Passive	Passive – Section 4	Segment/Component	Not Allowed	No

Notes:
[1] The assessments of the qualitative considerations are agreed upon by the IDP in accordance with Section 9.2. In some cases, a 50.69 categorization team may provide preliminary assessments of the seven considerations for the IDP's consideration, however the final assessments of the seven considerations are the direct responsibility of the IDP.

The seven considerations are addressed preliminarily by the 50.69 categorization team for at least the system functions that are not found to be HSS due to any other categorization step. Each of the seven considerations requires a supporting justification for confirming (true response) or not confirming (false response) that consideration. If the 50.69 categorization team determines that one or more of the seven considerations cannot be confirmed, then that function is presented to the IDP as preliminary HSS. Conversely, if all the seven considerations are confirmed, then the function is presented to the IDP as preliminary LSS.

The System Categorization Document, including the justifications provided for the qualitative considerations, is reviewed by the IDP. The IDP is responsible for reviewing the preliminary assessment to the same level of detail as the 50.69 team (i.e. all considerations for all functions are reviewed). The IDP may confirm the preliminary function risk and associated justification or may direct that it be changed based upon their expert knowledge. Because the Qualitative Criteria are the direct responsibility of the IDP, changes may be made from preliminary HSS to LSS or from preliminary LSS to HSS at the discretion of the IDP. If the IDP determines any of the seven considerations cannot be confirmed (false response) for a function, then the final categorization of that function is HSS.

[2] IDP consideration of seismic insights can also result in an LSS to HSS determination.

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). Except for seismic, 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. Components having seismic functions may be HSS or LSS based on the IDP's consideration of the seismic insights applicable to the system being categorized. Therefore, if an HSS component is mapped to an 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. For the seismic hazard, given that FNP is a seismic Tier 1 (low seismic hazard) plant as defined in Reference [3], seismic considerations are not required to drive an HSS determination at the component level, but the IDP will consider available seismic information pertinent to the components being categorized and can, at its discretion, determine that a component should be HSS based on that information.

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

- The IDP will be composed of a group of at least five experts who collectively have expertise in plant operation, design (mechanical and electrical) engineering, system engineering, safety analysis, and 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 SNC procedures.

- Decisions of the IDP will be arrived at by consensus. Differing opinions will be documented and resolved, if possible. However, a simple majority of the panel is sufficient for final decisions regarding safety significant and LSS.

- 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 SE (Reference [4]) 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 LSS.

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

- FNP proposes to apply an alternative seismic approach to those listed in NEI 00-04 Sections 1.5 and 5.3. This approach is specified in EPRI 3002017583 (Reference [3]) for Tier 1 plants and is discussed in Section 3.2.3.

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

- Internal Event Risks: Internal events including internal flooding PRA model uses common backbone model Revision 10 Version 3 (refer to Attachment 2).

- Fire Risks: Fire PRA model uses common backbone model Revision 10 Version 3 (refer to Attachment 2).

- Seismic Risks: EPRI Alternative Approach in EPRI 3002017583 (Reference [3]) for Tier 1 plants with the additional considerations discussed in Section 3.2.3 of this LAR.

- Other External Risks (e.g., tornados, external floods): An evaluation of external hazards was performed using Part 6 of the ASME/ANS PRA Standard (Reference [5]). The other external hazards were determined to be insignificant contributors to plant risk.

- Low Power and Shutdown Risks: Qualitative defense-in-depth (DID) shutdown model for shutdown Configuration Risk Management (CRM) based on the framework for DID provided in NUMARC 91-06, "Guidance for Industry Actions to Assess Shutdown Management" [6], 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 periodic 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 [7] (ML090930246) consistent with the related Safety Evaluation (SE) 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 [4]). 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 by the NRC in the ANO2-R&R-004 for the passive categorization of Class 2, 3, and non-class components. This is the same passive SSC scope the NRC has conditionally endorsed in ASME Code Cases N-660 and N-662 as published in Regulatory Guide 1.147, Revision 15. Both code cases employ a similar risk-informed safety classification of SSCs in order to change the repair/ replacement requirements of the affected LSS components. All ASME Code Class 1 SSCs with a pressure retaining function, as well as supports, will be assigned high safety-significant, HSS, for passive categorization which will result in HSS for its risk-

informed safety classification and cannot be changed by the IDP. Therefore, this methodology and scope for passive categorization is acceptable and appropriate for use at FNP 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 and there are no PRA upgrades that have not been peer reviewed.

3.2.1 Internal Events and Internal Flooding

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

3.2.2 Fire Hazards

The FNP categorization process for fire hazards will use a peer reviewed plant-specific fire PRA model. The internal Fire PRA model was developed consistent with NUREG/CR-6850 and only utilizes methods previously accepted by the NRC. The SNC risk management process ensures that the PRA model used in this application reflects the as-built and as-operated plant for FNP. Attachment 2 at the end of this enclosure identifies the applicable Fire PRA model.

3.2.3 Seismic Hazards

10 CFR 50.69(c)(1) requires the use of PRA to assess risk from internal events. For other risk hazards such as seismic, 10 CFR 50.69 (b)(2) allows, and NEI 00-04 (Reference [1]) summarizes, the use of other methods for determining SSC functional importance as part of an integrated, systematic process. For the FNP seismic hazard assessment, FNP proposes to use a risk informed graded approach that meets the requirements of 10 CFR 50.69 (b)(2) as an alternative to those listed in NEI 00-04 sections 1.5 and 5.3. This approach is specified in Electric Power Research Institute (EPRI) 3002017583, “Alternative Approaches for Addressing Seismic Risk in 10 CFR 50.69 Risk-Informed Categorization,” Reference [3], and includes additional qualitative considerations that are discussed in this section.

FNP meets the EPRI 3002017583 Tier 1 criteria for a “Low Seismic Hazard/High Seismic Margin” site. The Tier 1 criteria are as follows:

> *“Tier 1: Plants where the GMRS [Ground Motion Response Spectrum] peak acceleration is at or below approximately 0.2g or where the GMRS is below or approximately equal to the SSE [Safe Shutdown Earthquake] between 1.0 Hz and 10 Hz. Examples are shown in Figures 2-1 and 2-2. At these sites, the GMRS is either very low or within the range of the SSE such that unique seismic categorization insights are not expected.”*

Note: EPRI 3002017583 applies to the Tier 1 sites in its entirety except for sections 2.3 (Tier 2 sites), 2.4 (Tier 3 sites), Appendix A (seismic correlation), and Appendix B (criteria for capacity-based screening).

The Tier 1 criterion (i.e. basis) in EPRI 3002017583 is a comparison of the ground motion response spectrum (GMRS, derived from the seismic hazard) to the safe shutdown earthquake (SSE, i.e., seismic design basis capability). U.S. nuclear power plants that utilize the 10 CFR 50.69 Seismic Alternative (EPRI 3002017583) will continue to compare GMRS to SSE.

The trial studies in EPRI 3002017583 show that seismic categorization insights are overlaid by other risk insights even at plants where the GMRS is far beyond the seismic design basis. Therefore, the basis for the Tier 1 classification and resulting criteria is not that the design basis insights are adequate. Instead, it is that consideration of the full range of the seismic hazard produces limited unique insights to the categorization process. That is the basis for the following statements in Table 4-1 of the EPRI report.

> *"At Tier 1 sites, the likelihood of identifying a unique seismic condition that would cause an SSC to be designated HSS is very low.*
>
> *Therefore, with little to no anticipated unique seismic insights, the 50.69 categorization process using the FPIE PRA and other risk evaluations along with the required Defense-in-Depth and IDP qualitative considerations are expected to adequately identify the safety-significant functions and SSCs required for those functions and no additional seismic reviews are necessary for 10 CFR 50.69 categorization."*

The proposed categorization approach for FNP is a risk-informed graded approach that is demonstrated to produce categorization insights equivalent to a seismic PRA. For Tier 1 plants, this approach relies on the insights gained from the seismic PRAs examined in Reference [3] along with confirmation that the site GMRS is low. Reference [3] demonstrates that seismic risk is adequately addressed for Tier 1 sites by the results of additional qualitative assessments discussed in this section and existing elements of the 10 CFR 50.69 categorization process specified in NEI 00-04.

For example, the 10 CFR 50.69 categorization process as defined in NEI 00-04 includes an Integral Assessment that weighs the hazard-specific relative importance of a component (e.g., internal events, internal fire, seismic) by the fraction of the total Core Damage Frequency (CDF) contributed by that hazard. The risk from an external hazard can be reduced from the default condition of HSS if the integral assessment meets the importance measure criteria for LSS. For Tier 1 sites, the seismic risk (CDF/LERF) will be low such that seismic hazard risk is unlikely to influence an HSS decision. In applying the EPRI 3002017583 process for Tier 1 sites to the FNP 10 CFR 50.69 categorization process, the IDP will be provided with the rationale for applying the EPRI 3002017583 guidance and informed of plant SSC-specific seismic insights for their consideration in the HSS/LSS deliberations.

EPRI 3002017583 recommends a risk-informed graded approach for addressing the seismic hazard in the 10 CFR 50.69 categorization process. There are a number of seismic fragility fundamental concepts that support a graded approach and there are important characteristics about the comparison of the seismic design basis (represented by the SSE) to the site-specific

seismic hazard (represented by the GMRS) that support the selected thresholds between the three evaluation Tiers in the EPRI report. The coupling of these concepts with the categorization process in NEI 00-04 are the key elements of the approach defined in EPRI 3002012998 for identifying unique seismic insights.

The seismic fragility of an SSC is a function of the margin between an SSC's seismic capacity and the site-specific seismic demand. References such as EPRI NP-6041 (Reference [8]) provide inherent seismic capacities for most SSCs that are not directly related to the site-specific seismic demand. This inherent seismic capacity is based on the non-seismic design loads (pressure, thermal, dead weight, etc.) and the required functions for the SSC. For example, a pump has a relatively high inherent seismic capacity based on its design; that same seismic capacity applies at a site with a very low demand and at a site with a very high demand. At sites with lower seismic demands such as FNP, it is not necessary to perform more detailed evaluations to demonstrate the inherent seismic capacities documented in industry sources such as Reference [8]. Low seismic demand sites have lower likelihood of seismically-induced failures and lesser challenges to plant systems. This, therefore, provides the technical basis for allowing use of a graded approach for addressing seismic hazard at FNP.

There are some plant features such as equipment anchorage that have seismic capacities more closely associated with the site-specific seismic demand since those specific features are specifically designed to meet that demand. However, even for these features, the design basis criteria have intended conservatisms that result in significant seismic margins within SSCs. These conservatisms are reflected in key aspects of the seismic design process. The SSCs used in nuclear power plants are intentionally designed using conservative methods and criteria to ensure that they have margins well above the required design bases. Experience has shown that design practices result in margins to realistic seismic capacities of 1.5 or more.

The following provides the basis for establishing Tier 1 criteria in EPRI 3002017583.

a. SSCs for which the inherent seismic capacities are applicable, or which are designed to the plant SSE will have low probabilities of failure at sites where the peak spectral acceleration of the GMRS < 0.2g or where the GMRS < SSE between 1 and 10 Hz.

b. The low probabilities of failure of individual components would also apply to components considered to have correlated seismic failures.

c. These low probabilities of failure lead to low seismic CDF and LERF estimates, from an absolute risk perspective.

d. The low seismic CDF and LERF estimates lead to reasonable confidence that seismic risk contributions would allow reducing a HSS to LSS due to the 10 CFR 50.69 Integral Assessment if the equipment is HSS only due to seismic considerations.

Test cases described in Section 3 of Reference [3] showed that it would be unusual even for moderate hazard plants to exhibit any unique seismic insights, including due to correlated failures. The plant specific Reference [3] test case information SNC is using and being incorporated by Reference into this application is described in Case Study A (Reference [9]), Case Study C (References [10], [11], and Case Study D (References [12], [13], [14]). Hence, while it is prudent to perform additional evaluations to identify conditions where correlated

failures may occur for Tier 2 sites, for Tier 1 sites such as FNP, correlation studies would not lead to new seismic insights or affect the baseline seismic CDF in any significant way.

The Tier 1 to Tier 2 threshold as defined in EPRI 3002017583 provides a clear and traceable boundary that can be consistently applied plant site to plant site. Additionally, because the boundary is well defined, if new information is obtained on the site hazard, a site's location within a particular Tier can be readily confirmed. In the unlikely event that the FNP seismic hazard changes to medium risk (i.e., Tier 2) at some future time, FNP will follow its categorization review and adjustment process procedures to review the changes to the plant and update, as appropriate, the SSC categorization in accordance with 10 CFR 50.69(e).

The following provides the basis for concluding that FNP meets the Tier 1 site criteria.

In response to the NRC 50.54(f) letter associated with post-Fukushima recommendations (Reference [15]), FNP submitted a seismic hazard screening report (Reference [16]) to the NRC. The maximum GMRS value for FNP 1-10 Hz range meets the Tier 1 criterion of approximately 0.2g in Reference [3]. The FNP SSE and GMRS curves from the seismic hazard and screening response in Reference [16]) are plotted in Figure 1 at the end of Attachment 4.

The NRC's staff assessment of the FNP seismic hazard and screening response is documented in Reference [17]. In section 3.4 of Reference [17], the NRC concluded that the methodology used by SNC in determining the GMRS was acceptable and that the GMRS determined by SNC adequately characterizes the reevaluated hazard for the FNP site.

Section 1.1.3 of Reference [3] cites various post-Fukushima seismic reviews performed for the U.S. fleet of nuclear power plants. For FNP, the specific seismic reviews prepared by the licensee and the NRC's staff assessments are provided here. These licensee documents were submitted under oath and affirmation to the NRC.

1. NTTF Recommendation 2.1 seismic hazard screening (References [15]), [17]).
2. NTTF Recommendation 2.3 seismic walkdowns (References [18], [19], [20], [21])
3. NTTF Recommendation 4.2 seismic mitigation strategy assessment (S-MSA) (References [22], [23])

The small percentage contribution of seismic to total plant risk makes it unlikely that an integral importance assessment for a component, as defined in NEI 00-04, would result in an overall HSS determination. Further, the low hazard relative to plant seismic capability makes it unlikely that any unique seismic condition would exist that would cause an SSC to be designated HSS for a Tier 1 site such as FNP.

As an enhancement to the EPRI study results as they pertain to FNP, the proposed FNP categorization approach for seismic hazards will include qualitative consideration of the mitigation capabilities of SSCs during seismically-induced events and seismic failure modes, based on insights obtained from prior seismic evaluations performed for FNP. For example, as part of the categorization team's preparation of the System Categorization Document (SCD) that is presented to the IDP, a section will be included in the SCD that summarizes the identified plant seismic insights pertinent to the system being categorized, and will also state the basis for

applicability of the EPRI 3002017583 study and the bases for FNP being a Tier 1 plant. The discussion of the Tier 1 bases will include such factors as:

- The low seismic hazard for the plant, which is subject to periodic reconsideration as new information becomes available through industry evaluations; and
- The definition of Tier 1 in the EPRI study.

At several steps of the categorization process (e.g., as noted in Figure 3-1 and Table 3-1) the categorization team will consider the available seismic insights relative to the system being categorized and document their conclusions in the SCD. Integrated importance measures over all modeled hazards (i.e., internal events, including internal flooding, and internal fire) for FNP are calculated per Section 5.6 of NEI 00-04, and components for which these measures exceed the specified criteria are preliminary HSS which cannot be changed to LSS. For HSS SSCs uniquely identified by the FNP PRA models but having design-basis functions during seismic events or functions credited for mitigation and prevention of severe accidents caused by seismic events, these will be addressed using non-PRA based qualitative assessments in conjunction with any seismic insights provided by the PRA.

For components that are HSS due to fire PRA but not HSS due to internal events PRA, the categorization team will review design-basis functions during seismic events or functions credited for mitigation and prevention of severe accidents caused by seismic events and characterize these for presentation to the IDP as additional qualitative inputs, which will also be described in the SCD.

The categorization team will review available FNP plant-specific seismic reviews and other resources such as those identified above. The objective is to identify plant-specific seismic insights derived from the above sources, relevant to the components in the system being categorized, that might include potentially important impacts such as:

- Impact of relay chatter
- Implications related to potential seismic interactions such as with block walls
- Seismic failures of passive SSCs such as tanks and heat exchangers
- Any known structural or anchorage issues with a particular SSC
- Components that are implicitly part of PRA-modeled functions (including relays)
- Components that may be subject to correlated failures

Such impacts would be compiled on an SSC basis. As each system is categorized, the system-specific seismic insights will be provided to the IDP for consideration as part of the IDP review process, as noted in Figure 3-1. As such, the IDP can challenge, from a seismic perspective, any candidate LSS recommendation for any SSC if they believe there is basis for doing so. Any decision by the IDP to lower preliminary HSS components to LSS will also consider the applicable seismic insights in that decision. These insights will provide the IDP a means to consider potential impacts of seismic events in the categorization process.

Use of the EPRI approach outlined in Reference [3] to assess seismic hazard risk for 10 CFR 50.69 with the additional reviews discussed above will provide a process for categorization of RISC–1, RISC–2, RISC–3, and RISC–4 SSCs that satisfies the requirements of § 50.69(c)..

Based on the above, the Summary/Conclusion/Recommendation from Section 2.2.3 of Reference [3] applies to FNP, i.e., FNP is a Tier 1 plant for which the GMRS is very low such that unique seismic categorization insights are expected to be minimal. As discussed in Reference [3], the likelihood of identifying a unique seismic insight that would cause an SSC to be designated HSS is very low. Therefore, with little to no anticipated unique seismic insights, the 10 CFR 50.69 categorization process using the FPIE PRA and other risk evaluations along with the defense-in-depth and qualitative assessment by the IDP adequately identify the safety-significant functions and SSCs.

3.2.4 Other External Hazards

All external hazards, except for seismic, were screened for applicability to FNP per a plant-specific evaluation in accordance with GL 88-20 (Reference [24]) and updated to use the criteria in ASME PRA Standard RA-Sa-2009. Attachment 4 provides a summary of the external hazards screening results. Attachment 5 provides a summary of the progressive screening approach for external hazards.

3.2.5 Low Power & Shutdown

Consistent with NEI 00-04, the FNP categorization process will use the shutdown safety management plan described in NUMARC 91-06 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 either of 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 SNC risk management process ensures that the applicable PRA models used in this application continues to reflect the as-built and as-operated plant for FNP. 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, SNC 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 Section 8 of NEI 00-04 and in the prescribed sensitivity studies discussed in Section 5.

In the overall risk sensitivity studies, SNC will utilize a factor of 3 to increase the unavailability or unreliability of LSS components consistent with that approved for Vogtle in Reference [4]. Consistent with the NEI 00-04 guidance, SNC 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 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.

The detailed process of identifying, characterizing and qualitative screening of model uncertainties is found in Section 5.3 of NUREG-1855 and Section 3.1.1 of EPRI TR-1016737 (Reference [25]). 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 FNP 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 risk calculations were considered key for this application.

Key FNP PRA model specific assumptions and sources of uncertainty for this application were identified and dispositioned in Attachment 6. The conclusion of this review is that no additional sensitivity analyses are required to address FNP 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 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 [26]), consistent with NRC RIS 2007-06.

The full scope internal events (including internal flooding) PRA peer review was subject to a self-assessment and a full scope peer review conducted in March 2010 against ASME/ANS RA-Sa-2009, Addenda A to ASME/ANS RA-S-2008 Standard for Level 1/Large Early Release Frequency Probabilistic Risk Assessment for Nuclear Power Plant Applications, ASME and the American Nuclear Society, December 2008 (Reference [5]), and any clarifications and qualifications provided in the NRC endorsement of the Standard contained in Revision 2 to RG 1.200, An Approach for Determining the Technical Adequacy of Probabilistic Risk Assessment Results for Risk-Informed Activities, March 2009 (Reference [26]).

A finding closure review was conducted on the internal events (including internal flooding) in October 2018. Closed findings were reviewed and closed using the process documented in Appendix X to NEI 05-04, NEI 07-12 and NEI 12-13, “Close-out of Facts and Observations” (F&Os) (Reference [27]) as accepted by NRC in the letter dated May 3, 2017 (ML17079A427) (Reference [28]). The results of this review have been documented and are available for NRC audit. All Finding level F&Os were closed out.

The fire PRA (FPRA) was subject to a self-assessment and a full scope peer review conducted in October 2011 by the Pressurized Water Reactors Owners Group (PWROG) against all technical elements in Section 4 of ASME/ANS RA-Sa-2009 (Reference [5]) and any clarifications and qualifications provided in RG 1.200, Revision 2 (Reference [26]). The results of this peer review were documented in the Peer Review Report (transmitted by Letter LTR-RAM-II-12-007, dated March 2012 (Reference [29]). After the full scope peer review, two focused-scope peer reviews (February 2018 and July 2018) and one Appendix X closure review (September 2018) were held. In the September 2018 closure review, all F&Os from the October 2011 full scope peer review, all F&Os from the February 2018 focused-scope peer review, and all F&Os from the July 2018 focused scope peer review were closed out. The results of this review have been documented and are available for NRC audit.

In November-December 2019, a focused-scope peer review of the Farley internal events, internal flooding, fire, and seismic PRAs against applicable requirements of the ASME/ANS PRA standard (Reference [5]) was conducted of the following PRA model upgrades:

- Reactor Coolant Pump Shutdown Seal Model (applicable to both internal events and internal flooding PRA models)
- Main Control Room Abandonment (applicable to fire PRA model only)
- FLEX modeling with FLEX HRA (applicable to all PRA models)

The FLEX HRA portion of the review was performed using the Integrated Human Event Analysis System (IDHEAS) Method based on Electric Power Research Institute (EPRI) FLEX Human Reliability Analysis (HRA) Report 3002013018 (Reference [30]). As a result of this focused-scope peer review, a total of seven F&Os were generated, all of which were "Suggestion" type F&Os.

The 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 FNP 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.

3.5 FEEDBACK AND ADJUSTMENT PROCESS

If significant changes to the plant risk profile are identified, or if it is identified that a RISC-3 or RISC-4 SSC can (or actually did) prevent a safety significant function from being satisfied, an immediate evaluation and review will be performed prior to the normally scheduled periodic review. Otherwise, the assessment of potential equipment performance changes and new technical information will be performed during the normally scheduled periodic review cycle.

To more specifically address the feedback and adjustment (i.e., performance monitoring) process as it pertains to the proposed FNP Tier 1 approach discussed in section 3.2.3, implementation of the SNC design control and corrective action programs will ensure the inputs for the qualitative determinations for seismic continue to remain valid to maintain compliance with the requirements of 10 CFR 50.69(e).

The performance monitoring process is described in SNC's 10 CFR 50.69 program documents. The program requires that the periodic review assess changes that could impact the categorization results and provides the Integrated Decision-making Panel (IDP) with an opportunity to recommend categorization and treatment adjustments. Station personnel from engineering, operations, risk management, regulatory affairs, and others have responsibilities for preparing and conducting various performance monitoring tasks that feed into this process. The intent of the performance monitoring reviews is to discover trends in component reliability; to identify and reverse negative performance trends and take corrective action if necessary.

The SNC configuration control process ensures that changes to the plant, including a physical change to the plant and changes to documents, are evaluated to determine the impact to drawings, design bases, licensing documents, programs, procedures, and training. The configuration control program has been updated to include a checklist of configuration activities to recognize those systems that have been categorized in accordance with 10 CFR 50.69, to ensure that any physical change to the plant or change to plant documents is evaluated prior to implementing those changes. The checklist includes:

- A review of the impact on the System Categorization Document (SCD) for configuration changes that may impact a categorized system under 10 CFR 50.69.

- Steps to be performed if redundancy, diversity, or separation requirements are identified or affected. These steps include identifying any potential seismic interaction between added or modified components and new or existing safety related or safe shutdown components or structures.

- Review of seismic dynamic qualification of components if the configuration change adds, relocates, or alters Seismic Category I mechanical or electrical components.

SNC has a comprehensive problem identification and corrective action program that ensures that issues are identified and resolved. Any issue that may impact the 10 CFR 50.69 categorization process will be identified and addressed through the problem identification and corrective action program, including seismic-related issues.

The SNC 10 CFR 50.69 program requires that SCDs cannot be approved by the IDP until the panel's comments have been resolved to the satisfaction of the IDP. This includes issues related to system-specific seismic insights considered by the IDP during categorization.

Scheduled periodic reviews at least once every other Unit 1 refueling outage will evaluate new insights resulting from available risk information (i.e., PRA model or other analysis used in the categorization) changes, design changes, operational changes, and SSC performance. If it is determined that these changes have affected the risk information or other elements of the categorization process such that the categorization results are more than minimally affected,

then the risk information and the categorization process will be updated. This scheduled review will include:

- A review of plant modifications since the last review that could impact the SSC categorization.
- A review of plant specific operating experience that could impact the SSC categorization.
- A review of the impact of the updated risk information on the categorization process results.
- A review of the importance measures used for screening in the categorization process.
- An update of the risk sensitivity study performed for the categorization.

In addition to the normally scheduled periodic reviews, if a PRA model or other risk information is upgraded, a review of the SSC categorization will be performed.

The periodic monitoring requirements of the 10 CFR 50.69 process will ensure that these issues are captured and addressed at a frequency commensurate with the issue severity. The 10 CFR 50.69 periodic monitoring program includes immediate and periodic reviews, that include the requirements of the regulation, to ensure that all issues that could affect 10 CFR 50.69 categorization are addressed. The periodic monitoring process also monitors the performance and condition of categorized SSCs to ensure that the assumptions for reliability in the categorization process are maintained.

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 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."

- 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.

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

4.2 NO SIGNIFICANT HAZARDS CONSIDERATION ANALYSIS

SNC 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.

SNC 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 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, SNC 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, July 2005.

[2] NRC Regulatory Guide 1.201, "Guidelines for Categorizing Structures, Systems, and Components in Nuclear Power Plants According to their Safety Significance," May 2006.

[3] Electric Power Research Institute (EPRI) 3002017583, Alternative Approaches for Addressing Seismic Risk in 10 CFR 50.69 Risk-Informed Categorization, Technical Update, February 2020.

[4] Vogtle Electric Generating Plant, Units 1 and 2 -Issuance of Amendments Re: Use of 10 CFR 50.69 (TAC NOS. ME9472 AND ME9473), December 17, 2014.

[5] ASME/ANS RA-S-2009, Addenda to ASME/ANS RA-S-2008, Standard for Level 1/Large Early Release Frequency Probabilistic Risk Assessment for Nuclear Power Plant Applications, February 2009.

[6] NUMARC 91-06, "Guidelines for Industry Actions to Assess Shutdown Management," December 1991.

[7] ANO SER Arkansas Nuclear One, Unit 2 - Approval of Request for Alternative AN02-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) (ML090930246), April 22, 2009.

[8] Electric Power Research Institute (EPRI) NP-6041-SL, "A Methodology for Assessment of Nuclear Plant Seismic Margin," Revision 1, August 1991.

[9] Peach Bottom Atomic Power Station Seismic Probabilistic Risk Assessment Report, "Response to NRC Request Regarding Recommendation 2.1 of the Near Term Task Force Review of Insights from the Fukushima Dai-ichi Accident," August 28, 2018 (ML18240A065).

[10] Plant C, License Amendment Request to Modify Approved 10 CFR 50.69 Categorization Process, June 22, 2017 (ML17173A875).

[11] Plant C, "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)," August 10, 2018 (ML18180A062).

[12] Seismic Probabilistic Risk Assessment for Plant D Nuclear Plant, Units 1 and 2, "Response to NRC Request for Information Pursuant to 10 CFR 50.54(f) Regarding Recommendation 2.1 of the NTTF Review of Insights from the Fukushima Dai-ichi Accident," June 30, 2017 (ML1718A485).

[13] Plant D Nuclear Plant Seismic Probabilistic Risk Assessment Supplemental Information, April 10, 2018 (ML18100A966).

[14] Plant D 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," November 29, 2018 (ML18334A363).

[15] U.S. Nuclear Regulatory Commission, Request for Information Pursuant to Title 10 of the Code of Federal Regulations 50.54(F) Regarding Recommendations 2.1,2.3, And 9.3, of the Near-Term Task Force Review of Insights from the Fukushima Dai-Ichi Accident, March 12, 2012 (ML12053A340).

[16] Joseph M. Farley Nuclear Plant - Units 1 and 2, Seismic Hazard and Screening Report for CEUS Sites, March 31, 2014 (ML14092A020) .

[17] Joseph M. Farley Nuclear Plant, Units 1 and 2- Staff Assessment of Information Provided Pursuant to Title 10 of the Code Of Federal Regulations Part 50, Section 50.54(F), "Seismic Hazard Reevaluations for Recommendation 2.1 of the Near-Term Task Force Review of Insights from the Fukushima Dal-Ichi Accident (TAC NOS. MF3832 AND MF3833)," October 16, 2015 (ML15287A092).

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[38] Joseph M. Farley Nuclear Plant - Units 1 and 2, "Seismic Hazard and Screening Report for CEUS Sites," March 31, 2014 (ML14092A020).

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[49] Electric Power Research Institute (EPRI) Technical Report TR-1026511, "Practical Guidance on the Use of PRA in Risk-Informed Applications with a Focus on the Treatment of Uncertainty," December 2012.

Attachment 1: List of Categorization Prerequisites

Southern Nuclear Operating Company (SNC) has existing fleet procedures which outline the process for categorization of plant systems. The SNC fleet procedures contain the elements/steps listed below for categorizing systems at FNP.

- Integrated Decision-Making Panel (IDP) member qualification requirements
- Qualitative assessment of system functions. System functions are qualitatively categorized as preliminary High Safety Significant (HSS) or Low Safety Significant (LSS) based on the seven criteria in Section 9 of NEI 00-04 (see Section 3.2). 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 Probabilistic Risk Assessment (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. Safety-related 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 core damage frequency (CDF) and large early release frequency (LERF) and meets the acceptance guidelines of Regulatory Guide 1.174.
- 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 the enclosure

Attachment 2: Description of PRA Models Used in Categorization

Hazard	Station	Unit	Model	Baseline CDF	Baseline LERF	Comments
Internal Events including Internal Flooding	FNP	1	Rev 10 V3 CBM	6.9E-6	3.6E-8	F-RIE-IEIF-U01, Version 5
Internal Events including Internal Flooding	FNP	2	Rev 10 V3 CBM	7.0E-6	3.7E-8	F-RIE-IEIF-U02, Version 5
Fire	FNP	1	Rev 10 V3 CBM	7.7E-5	2.7E-6	F-RIE-FIREPRA-U00-014
Fire	FNP	2	Rev 10 V3 CBM	7.7E-5	5.2E-6	F-RIE-FIREPRA-U00-014

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

Finding Number	Supporting Requirement(s)	Capability Category (CC)	Description	Disposition for 50.69
This attachment is intentionally blank. There are no open peer-review findings and self-assessment open items for the Internal events / Internal flooding or Fire PRA models.				

Attachment 4: External Hazards Screening

External Hazard	Screening Result		
	Screened? (Y/N)	Screening Criterion (Note a)	**Comment**
Aircraft Impact	Y	PS4	There are no airports within 10 miles of the plant. There are no military facilities or military training routes close to the plant. Aircraft hazard is not a design basis hazard event for the plant and the UFSAR (Reference [31]) using the most recent data confirms this conclusion. As a result, beyond design basis challenges from accidental aircraft impacts are screened out. (Reference [32]).
Avalanche	Y	C3	Topography is such that no avalanche is possible as plant is not located near large mountains where snow avalanches are prevalent.
Biological Events	Y	C1 C5	The accumulation or deposition of vegetation or organisms (e.g. zebra mussels, clams, fish) on an intake structure or internal to a system that uses an intake structure would not occur as the Chattahoochee River is not the Ultimate Heat Sink (UHS) for FNP (**C1**). The Service Water Storage Pond provides this service. As this is slow to develop, there would be adequate warning for these events (**C5**).
Coastal Erosion	Y	C3	FNP is a riverine site located inland.
Drought	Y	C1 C5	Drought is a slowly developing hazard (**C5**). The plant location (riverine site with upstream dams: Walter F. George Dam and Columbia Lock and Dam; and downstream dam, Jim Woodruff Dam) precludes impact on FNP. The storage pond is capable of providing sufficient cooling for at least 30 days (**C1**).

External Hazard	Screening Result		
	Screened? (Y/N)	Screening Criterion (Note a)	**Comment**
External Flooding	Y	C1	The NRC previously approved the FNP External Flooding hazard screening analysis as discussed in its safety evaluation for Amendments Nos. 225 and 222 to permit the use of Risk-Informed Completion Times (RICTs) (Reference [33]). The results of that screening analysis also apply to this 50.69 application. In summary, the external flooding mechanisms besides LIP and combined effects flooding either do not pose a challenge to the plant or do not apply to the plant. Regarding LIP, the doors in Table A-1 of FNP's RAI 11.a response (Reference [34]) are credited for screening the LIP flood mechanism. Categorization of equipment credited for screening of hazards will follow the screening process shown in Figure 5-6 of NEI 00-04 (Reference [1]). Regarding the combined effects flooding mechanism, per Reference [34], the maximum still flood elevation is below the finished floor elevation of the plant with credit taken for the Kontek Vehicle Barrier System (VBS) to prevent wave action from propagation to the plant's safety-related buildings because the maximum flooding against the VBS is below the grade of the power block. Categorization of the VBS will follow the screening process shown in Figure 5-6 of NEI 00-04 (Reference [1]).
Extreme Wind or Tornado	Y	C1 PS4	The NRC previously approved the FNP Extreme Wind or Tornado hazard screening analysis as discussed in its safety evaluation for Amendments Nos. 225 and 222 to permit the use of Risk-Informed Completion Times (RICTs) (Reference [33]). The results of the screening analysis also apply to this 50.69 application.

External Hazard	Screening Result		
	Screened? (Y/N)	Screening Criterion (Note a)	**Comment**
			In summary, all above ground Category I structures required for safe shutdown are designed to withstand tornado loadings. Based on Section 3.3.2.1 of the FNP FSAR (Reference [31]) , the design includes (1) dynamic wind pressure associated with a tornado having a wind velocity of 300 mph and translational velocity of 60 mph, and (2) 3 psi pressure differential. Per Table 6-1 of NUREG/CR-4461, Rev. 2, the 10-7 annual probability tornado wind speed is 283 mph (i.e., less than the 300 mph design) for Joseph M. Farley, based on the more conservative F-scale; using the more recent EF-scale the 10-7 annual probability tornado wind speed is even lower (217 mph). Based on the plant design for wind pressure and the low frequency (<10-7/yr) of design tornadoes, a demonstrably conservative estimate of CDF associated with tornado hazards other than tornado missile is much less than 10E-6/yr. Regarding tornado missiles, a plant specific analysis was performed as described in FNP's RAI 11.f and 11.g response (Reference [34]). Per this analysis, the CDF associated with tornado missiles is estimated to be much less than 1 E-6/ yr. Please note that although FNP subsequently updated the licensing basis for tornado missile protection from TORMIS to the tornado missile risk evaluator (TMRE) methodology as described in NEI 17-02, the conclusions pertaining to tornado missile screening described in FNP's RAI 11.f and 11.g responses remain valid.
Fog	Y	C4	Water droplets suspended in the atmosphere at or near the Earth's surface that limit visibility affect the frequency of occurrence of other

External Hazard	Screening Result		
	Screened? (Y/N)	Screening Criterion (Note a)	**Comment**
			hazards (e.g. highway accidents, aircraft landing and take-off accidents) and is indirectly considered. Fog has a rare occurrence in the site region. Section 2.3.2.2 of UFSAR (Reference [31]) states that visibility of less than 1/4 mile occurs less than 1.3 percent of the time.
Forest or Range Fire	Y	C3	Per UFSAR Sec 2.2.3.6 (Reference [31]) wooded areas are sufficiently far from the plant structures that brush and forest fires do not present a hazard (Reference [32]).
Frost	Y	C1	Snow and Ice Cover hazards govern this risk.
Hail	Y	C1	Hail may occur but there are no openings in the walls or roofs of safety related buildings through which hail may enter and damage essential equipment. Tornado missile protection features, structural walls and roofs are adequate to withstand the impact of hail.
High Summer Temperature	Y	C2	The highest recorded temperature at Dothan Airport was 108°F. The HVAC systems are designed to maintain prescribed building temperatures during outside temperature variations between 20°F and 95°F (Reference [31]). Even if the maximum temperature exceeds the design limits for HVAC systems, such exceedance lasts only for a brief period and, given the thermal inertia of the concrete structures where safety-related equipment are located, will not have any impact.

External Hazard	Screening Result		
	Screened? (Y/N)	Screening Criterion (Note a)	**Comment**
High Tide, Lake Level, or River Stage	Y	C3	This event is of negligible impact on plant. The plant location (riverine with upstream and downstream dams) preclude impact on plant due to this hazard. See External Flooding discussion for more information.
Hurricane	Y	C1 C3	FNP is an inland site (**C3**). Hurricane wind effects are bounded by extreme winds and tornados assessment (**C1**). See the Extreme Wind or Tornado hazard.
Ice Cover	Y	C3	Icing does not normally occur on the Chattahoochee River at FNP. The only incidence of icing occurred in 1961 along the banks in slack water areas. No record of the river being iced over at this location has been found. Therefore, there would be no interference with the flow of water into the river water intake due to ice. Even if the surface did become frozen there would be no interference with withdrawal of water by the river water intake due to depth of water in the river (UFSAR Section 2.4.7 (Reference [31])).
Industrial or Military Facility Accident	Y	C1	Per UFSAR Section 2.2.1 (Reference [31]), there are no military bases or firing ranges, oil pipelines, or tank farms located within a 10-mile radius of the plant site.
Internal Flooding	N/A	N/A	The FNP Internal Events and Internal Flooding PRA model addresses risk from internal flooding events.
Internal Fire	N/A	N/A	The FNP Internal Fire PRA model addresses risk from internal fires.

External Hazard	Screening Result		
	Screened? (Y/N)	Screening Criterion (Note a)	**Comment**
Landslide	Y	C3	In the immediate vicinity of FNP there are no steep hills.
Lightning	Y	C1	Lightning strikes are not uncommon in nuclear plant experience. They can result in losses of off-site power or surges in instrumentation output if grounding is not fully effective. The latter events often lead to reactor trips. This was considered in plant design.
Low Lake Level or River Stage	Y	C1 C5	A decrease in the water level of the lake or river does not impact FNP as FNP does not rely on Chattahoochee River for the UHS (**C5**) since the storage pond provides the necessary UHS requirements (**C1**).
Low Winter Temperature	Y	C5	The lowest recorded temperature at Dothan Airport was 5°F; the plant design basis is 17°F. The HVAC systems are designed to maintain prescribed building temperatures during outside temperature variations between 20°F and 95°F (Reference [31]). Even if the minimum temperature exceeds the design limits for HVAC systems, such exceedance lasts only for a brief period and, given the thermal inertia of the concrete structures where safety-related equipment are located, will not have any impact. Therefore, the temperatures inside the plant buildings are expected to be higher than 17°F.
Meteorite or Satellite Impact	Y	PS4	This hazard is of negligible likelihood of impact to the site (very low event probability).
Pipeline Accident	Y	C1	A 6-in gas pipeline passes about 2.5 miles east of the main plant building. This is a grade B pipe

External Hazard	Screening Result		
	Screened? (Y/N)	Screening Criterion (Note a)	**Comment**
			with a nominal wall thickness of 0.188 in. and an average depth of 30 in. It carries 12 million cubic feet per day. In Section 2.2.3.2 of the UFSAR (Reference [31]), an explosion or fire following a break of this pipe would not be hazardous for FNP. Therefore, the hazard posed by pipeline accidents is screened out from the FNP PRA.
Release of Chemicals in Onsite Storage	Y	C4	Chemicals stored near FNP have been evaluated annually since the OL issuance. A comprehensive review of chemicals in onsite storage was evaluated and documented(Reference [32]). Procedures are in place to assess the impact of any new chemical procured for plant operations on control room habitability based on the toxicity limits given in RG 1.78 (Reference [35]) See also the Industrial or Military Facility Accident hazard.
River Diversion	Y	C3	UFSAR Section 2.4.9 (Reference [31]) states that the river upstream from the site does not have sufficiently high banks to cause a potential diversion of the river and bypass of the intake structure. With Lake Seminole varying between el 76 ft MSL and 78 ft MSL, a temporary blockage of the river upstream from FNP would not seriously affect the quantity of water available to the river water intake. Even if the river was temporarily blocked, cooling water could be obtained from the storage pond.

External Hazard	Screening Result		
	Screened? (Y/N)	Screening Criterion (Note a)	**Comment**
Sand or Dust Storm	Y	C3	A strong wind storm with airborne particles of sand and dust is not relevant for this region.
Seiche	Y	C3	There is no large body of water close to the site for this event.
Seismic Activity	N/A	N/A	See Section 3.2.3 and Figure A4-1 in this Attachment.
Snow	Y	C1	The 100 year snow load is estimated as 10 psf. The design basis roof live load for seismic Category I structures is at least 20 psf (Reference [32]). .
Soil Shrink-Swell Consolidation	Y	C1 C5	This is slow to develop and procedures are in place to monitor differential settlement (UFSAR Section 2B.7.3.1 (Reference [31])) (**C5**). All measured settlements are small. There have been no construction problems experienced due to total or differential settlement of foundations (**C1**).
Storm Surge	Y	C3	FNP is located inland and is not affected by storm surge.
Toxic Gas	Y	C4	Toxic gas is covered under these hazards: • Release of Chemicals in Onsite Storage • Industrial or Military Facility Accident • Transportation Accident

External Hazard	Screening Result		
	Screened? (Y/N)	Screening Criterion (Note a)	**Comment**
Transportation Accident	Y	C1 C3	Analysis of postulated accidents on nearby transportation routes has shown (Reference [32]) that they do not pose a credible threat to FNP since these routes are farther than the safe distances specified in RG 1.78 (Reference [35]) and RG 1.91 (Reference [36]) (**C1, C3**).
Tsunami	Y	C3	FNP is located inland and is not exposed to the Tsunami threat
Turbine-Generated Missiles	Y	C1 PS4	The probabilistic analysis performed for failures of turbines in Units 1 & 2 (Reference [32]) shows the probability of turbine missile damage is less than the NRC accepted value (per RG 1.115, Reference [37]) of 1x10-7 per year. To further reduce the probability of turbine failure, FNP has adopted a rigorous maintenance program. Therefore, given the worst case probability of turbine missile damage of 1E-7 the bounding CDF assuming a CCDP of 1.0 is less than 1E-6 per year (Reference [32]) (**C1**, **PS4**).
Volcanic Activity	Y	C3	Not applicable to the site because of location (no active or dormant volcanoes located near plant site).
Waves	Y	C3	FNP is located inland and is not affected by any wave activity.
Note a – See Attachment 5 for descriptions of the screening criteria.			

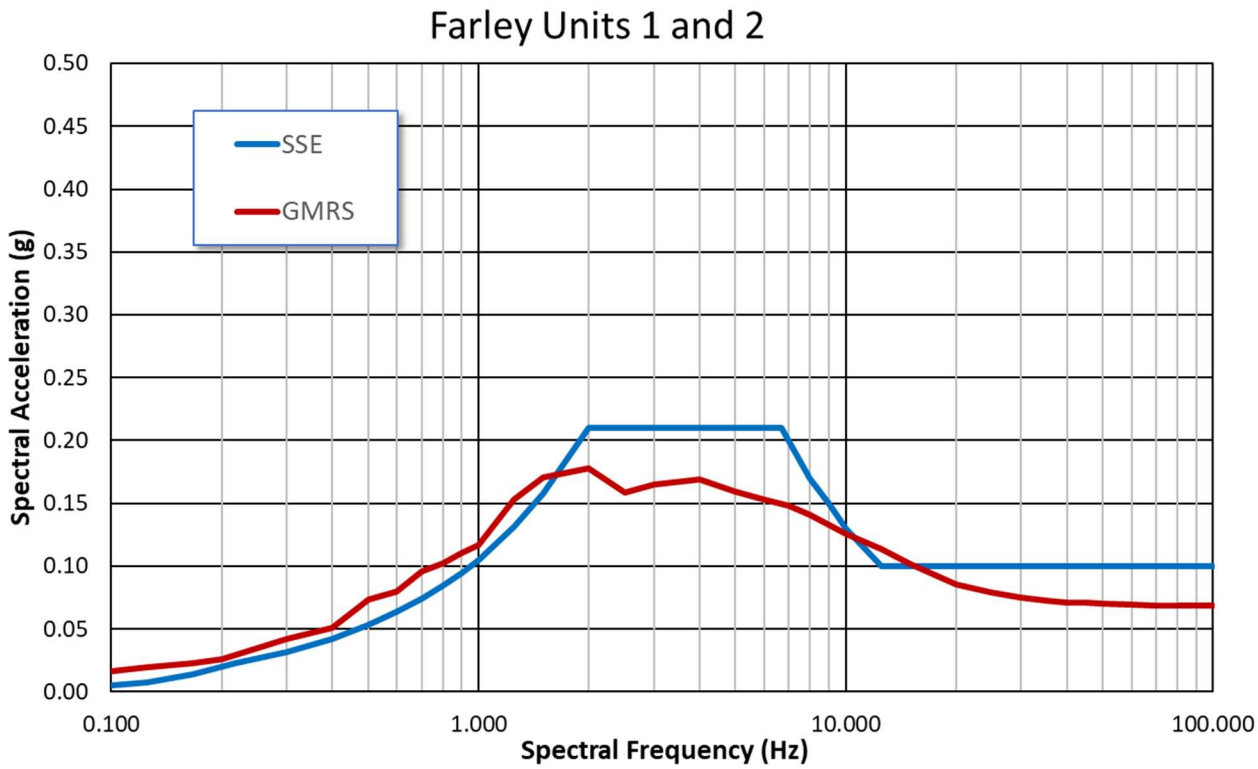

Figure A4-1: GMRS and SSE Response Spectra for FNP

(From FNP Seismic Evaluation Report (Reference [38]), Table 2.4-1 (GMRS) and Table 3.3-1 (SSE)

Attachment 5: Progressive Screening Approach for Addressing External Hazards

Event Analysis	Criterion	Source	Comments
Initial Preliminary Screening	C1. Event damage potential is < 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	

Event Analysis	Criterion	Source	Comments
	PS3. Design basis event mean frequency is < 1E-5/y 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-6/y.	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

Internal Events / Internal Flood (IEIF) PRA Model:

The process used to identify PRA model uncertainties and their impact is described in the Farley IEIFPRA uncertainty documentation (Reference [39]).

In the uncertainty documentation, EPRI 1016737 (Reference [25]) was used to provide guidance for a structured process for addressing uncertainties in PRA results in the context of risk-informed decision-making. Appendix A of EPRI 1016737 is used as a template to document plant-specific issue characterization and assessments to fully satisfy the related supporting requirements.

Although the uncertainty documentation does not explicitly refer to NUREG-1855 Revision 1, the uncertainty evaluation process and sources in EPRI 1016737 are addressed. The information in this Attachment for IEIFPRA uncertainty information was distilled from the IEIFPRA uncertainty documentation.

The evaluation of sources of uncertainty in the IEIFPRA as summarized in the table below considered the quantification/summary documentation performed for the recent Farley 4B application (Reference [40]) and subsequent RAI responses as pertains to this 50.69 application.

The parametric uncertainty analysis for the Farley IEIFPRA is documented in the quantification/model documentation. The parametric uncertainty analysis addresses the State-of-Knowledge Correlation (SOKC) by the use of system level type codes for basic events. This applies the same variability of all components of that type within a system during the analysis. The parametric uncertainty analyses in the PRA quantification model/model documentation demonstrate that the point estimate mean values provide a close representation of the propagated mean values reflecting SOKC and the propagated mean total CDF and LERF values were confirmed to meet RG 1.174 Revision 3 (Reference [41]).

The completeness uncertainty associated with scope and level of detail are documented in the PRA but are only considered for their impact on a specific application. No specific issues of PRA completeness have been identified relative to the 50.69 application, based on the results of the internal events PRA and fire PRA peer reviews.

Note: as part of the required 50.69 PRA categorization sensitivity cases directed by NEI 00-04 (Reference [1]), the IEIFPRA model human error and common cause basic events are increased to their 95th percentile and also decreased to their 5th percentile values; and maintenance unavailability terms are set to 0.0. These results are capable of driving a component and respective functions HSS. The uncertainty of the PRA modeled HEPs and CCFs are thus accounted for in the 50.69 application.

The table below describes the internal events / internal flooding (IE / IF) PRA sources of model uncertainty and their impact.

IE / IF PRA Sources of Assumption/ Uncertainty	IE / IF PRA 50.69 Impact	IE / IF PRA Model Sensitivity and Disposition (50.69)
The FNP HRA is based on the Unit 1 PRA model, which is applicable to Unit 2 as well.	Differences among the units that could relate to the HRA are those that stem from differences in specific system differences, configuration differences, procedures and training differences.	The configuration and system differences are accounted for in the development of system fault trees. As part of this development of the system fault trees, a comparison between units were done. There were no significant differences between the units that would affect the HRA. Configuration/system differences are noted in shared-unit procedures and in training material and would have been captured in the HRA analysis. This is not a key source of uncertainty and will not be an issue for the 50.69 application.
Room cooling failures	Room cooling failures are relatively small contributors to bus failure, even assuming no credit for operator response to a room cooling failure.	GOTHIC heat-up analysis determined which electrical equipment, including AC & DC Bus/Switchgears, are dependent upon successful operation of the associated room coolers during the 24-hour mission time modeled in the PRA. The failure of these room coolers was modeled in the applicable system fault trees. This is not a key source of uncertainty and will not be an issue for the 50.69 application.

IE / IF PRA Sources of Assumption/ Uncertainty	IE / IF PRA 50.69 Impact	IE / IF PRA Model Sensitivity and Disposition (50.69)
Plugging of the SW strainers	Plugging of the SW strainers is not included in the Loss of Service Water initiating event fault tree logic models. Plant experience shows that there has been no loss of SW induced by strainer plugging. The SW system is in continuous operation and the strainers are routinely (every 4 hours) backwashed by the operators. The differential pressure switch across each SW strainer inlet and outlet alarms on the SW structure alarm panel upon an increase in differential pressure. Any alarm on this panel causes a SW Structure Alarm annunciator in the main control room. However, the cause of the alarm can only be determined from the SW structure panel. The strainers are located downstream of the pumps. Therefore, any credible strainer plugging fault would likely fail the pumps first. In addition, the water in the wet pit has been screened in both the river water and service water intake screens	Strainer plugging would be expected to be a slow process, and if the strainers began to clog, the increased delta-P would be expected to be detected and corrected within a short period of time. However, since routine monitoring of the SWIS may be impacted following other initiating events, plugging of the strainers and failure of operator action to respond to the high differential pressure alarm is modeled in the mitigating system fault trees. This is not a key source of uncertainty and will not be an issue for the 50.69 application.
Credit for battery life out to two hours is based on conservative FSAR analyses without explicit	The two-hour battery life assumes procedurally directed load shedding has not been implemented.	The impact regarding battery lifetime is mitigated in part in the PRA model by requiring the battery charger for DC demands

IE / IF PRA Sources of Assumption/ Uncertainty	IE / IF PRA 50.69 Impact	IE / IF PRA Model Sensitivity and Disposition (50.69)
representation of or credit for successful load shedding	Without recovery of DC power at two hours, equipment requiring DC power (e.g., turbine-driven AFW pump (TDAFW)) is assumed unavailable after battery depletion.	beyond the assumed two-hour battery life (for other than station blackout scenarios). The impact of the battery lifetime is further constrained by the fact that the turbine driven auxiliary feedwater pump relies on instrument air operated steam admission valves. The steam admission valves fail close on a loss of air; however, an air reservoir is provided that will hold these valves open for a nominal two hours. Site emergency operating procedures direct the operators to take manual control of turbine driven auxiliary feedwater pump during this scenario. This is modeled in the PRA. The DC supply to the TDAFW pump has a four-hour rating and manual action could be taken to maintain the steam admission valves open beyond 2 hours (Reference [42]). Since successful operation of the Turbine-driven AFW pump relies on steam admission valves that are operated by instrument air and not DC power, the AFW system's reliance on DC power is reduced. This is not a key source of uncertainty and will not be an issue for the 50.69 application.

IE / IF PRA Sources of Assumption/ Uncertainty	IE / IF PRA 50.69 Impact	IE / IF PRA Model Sensitivity and Disposition (50.69)
Treatment of flow diversion paths	In the PRA model, diverted flow paths in fluid systems are removed if the cross-sectional area of the diversion path is less than ten percent of the cross-sectional area of the main process flow path, and potential flow diversion paths that are greater than one third (1/3) the diameter of the main flow path should be further evaluated.	This approach does not explicitly treat pressure effects of flow diversions from high pressure to low pressure, and no supporting thermal hydraulic analyses are performed to assess the validity of this assumption for these cases. This is not a key source of uncertainty and will not be an issue for the 50.69 application.
Interfacing Systems LOCA (ISLOCA) frequencies	A detailed ISLOCA analyses was performed that involved screening of potential ISLOCA pathways, calculation of the frequency of failure of the high pressure/low pressure interface of each unscreened interfacing system, and calculation of the probability of piping or component failure in the interfacing system as a result of the exposure to high pressure. Calculations were performed to assess the failure frequency of each scenario based on its specific configuration. These calculations are based on NSAC-154 (Reference [43]) and NUREG/CR-5102 (Reference [44]) with modifications as appropriate to represent differences in the Farley configuration.	The impacts of overpressure on each of the above ISLOCA scenario pathways were evaluated using the guidelines of NUREG/CR-5862 (Reference [45]). The approach for the ISLOCA frequency determination applies state-of-the art methods. This is not a key source of uncertainty and will not be an issue for the 50.69 application.
The use of a single value in the PRA model for unrecoverable failure due to	There is not a consistent method for the treatment of ECCS sump performance.	Unrecoverable failure of recirculation due to sump screen plugging is included in the model

IE / IF PRA Sources of Assumption/ Uncertainty	IE / IF PRA 50.69 Impact	IE / IF PRA Model Sensitivity and Disposition (50.69)
sump screen plugging for all sequences.		for each sump intake based on NUREG/CR- 4550 [46]. This is not a key source of uncertainty and will not be an issue for the 50.69 application.
Failure of core cooling following containment failure is not explicitly modeled.	A combination of generic and plant-specific analyses are used to evaluate the impact of containment failure on ECCS recirculation.	Since the Farley design basis does not credit containment overpressure in the RHR pump NPSH analysis, and the Farley PRA requires operable cooling through the RHR heat exchangers or containment fan coolers for success of ECCS recirculation, the loss of NPSH due to steam release from an unisolated containment is considered unlikely. This is not a key source of uncertainty and will not be an issue for the 50.69 application.
Human Error Probabilities (HEPs): Uncertainties associated with the assumptions and method of calculation of HEPs for the Human Reliability Analysis (HRA) may introduce uncertainty.	Detailed evaluations of HEPs are performed for the risk significant human failure events (HFEs) using industry consensus methods. Mean values are used for the modeled HEPs. Uncertainty associated with the mean values can have an impact on CDF and LERF results.	The FNP PRA model is based on industry consensus modeling approaches for its HEP calculations, so this is not considered a significant source of epistemic uncertainty. Additionally, for the 50.69 program, the guidance in NEI 00-04 (Reference [1]) specifies that certain sensitivity studies be conducted for each PRA model to address key sources of

IE / IF PRA Sources of Assumption/ Uncertainty	IE / IF PRA 50.69 Impact	IE / IF PRA Model Sensitivity and Disposition (50.69)
		uncertainty. The sensitivity studies are performed to ensure that assumptions and sources of uncertainty (e.g., human error, common cause failure, maintenance probabilities, and manual suppression probabilities for fire) do not mask the SSC(s) importance.

Disposition of Key Assumptions/Sources of Uncertainty

Fire PRA (FPRA) Model:

The process used to identify uncertainties and their impact is described for the fire PRA in the Farley FPRA uncertainty documentation (Reference [47]).

The evaluation examines sources of uncertainty for each of the FPRA development tasks defined in NUREG/CR-6850 (Reference [48]) and interpretations. Although this evaluation does not specifically refer to EPRI 1026511 (Reference [49]) and therefore does not specifically address each of the Fire PRA entries in Appendix B of that report, the considerations in the FPRA quantification/summary documentation are consistent with EPRI 1026511 Appendix B and represent an appropriate set of potential sources of uncertainty for this application.

In addition, the FPRA uncertainty documentation includes a discussion of sensitivities that were performed to evaluate the reasonableness of various modeling uncertainties in the FPRA model. As noted in the quantification/summary documentation, the FNP FPRA analysis is believed to represent a somewhat conservative estimation of fire risk, within the constraints of the requirements for a model acceptable for the NFPA-805 program.

The evaluation of sources of uncertainty in the FPRA as summarized in the table below considered the recent Farley 4B application (Reference [40]) and subsequent RAI responses as pertains to this 50.69 application.

The parametric uncertainty analysis for the Farley Fire PRA is provided in the uncertainty documentation. The parametric uncertainty analysis addresses the State-of-Knowledge Correlation (SOKC) by the use of system level type codes for basic events. This applies the same variability of all components of that type within a system during the analysis. The parametric uncertainty analyses in the PRA quantification model/model documentation demonstrate that the point estimate mean values provide a close representation of the propagated mean values reflecting SOKC and the propagated mean total CDF and LERF values were confirmed to meet RG 1.174 Revision 3 (Reference [41]).

The completeness uncertainty associated with scope and level of detail are documented in the PRA but are only considered for their impact on a specific application. No specific issues of PRA completeness have been identified relative to the 50.69 application, based on the results of the internal events PRA and fire PRA peer reviews.

As part of the required 50.69 PRA categorization sensitivity cases directed by NEI 00-04 (Reference [8]), the FPRA model human error and common cause basic events are increased to their 95th percentile and also decreased to their 5th percentile values; maintenance unavailability terms are set to 0.0; and there is no credit for manual suppression. These results are capable of driving a component and respective functions HSS and therefore the PRA modeled HEPs and CCFs are accounted for in the 50.69 application.

The table below describes the fire PRA sources of model uncertainty and their impact.

Fire PRA Description	Fire PRA Sources of Uncertainty	Fire PRA Disposition
Analysis Boundary and Partitioning	This task establishes the overall spatial scope of the analysis and provides a framework for organizing the data for the analysis. The partitioning features credited are required to satisfy established industry standards.	The methodology for the Analysis Boundary and Partitioning task does not introduce any epistemic uncertainties that would require sensitivity treatment.
Fire PRA Component Selection	This task involves the selection of components to be treated in the analysis in the context of initiating events and mitigation. The potential sources of uncertainty include those inherent in the internal events PRA model as that model provides the foundation for the FPRA.	In the context of the FPRA, the uncertainty that is unique to the analysis is related to initiating event identification. However, that impact is minimized though use of the PWROG Generic MSO list and the process used to identify and assess potential MSOs. The methodology for the Component Selection task does not introduce any epistemic uncertainties that would require sensitivity treatment.
Cable Selection	The selection of cables to be considered in the analysis is identified using industry guidance documents. Some systems are not credited in the FPRA and are therefore treated as being failed everywhere. The overall process is essentially the same as that used to perform the analyses to demonstrate compliance with 10 CFR 50.48.	As part of the Fire PRA development, there are components that are included in the basic event mapping process but are not included in the circuit analysis task. The basis for this is that these components are either not credited in the Fire PRA or they are of sufficiently low risk such that the overall change in risk is not impacted by the crediting of these components. These components make up a list referred to as "UNL" or unknown location. Uncertainty associated with this cable selection task was discussed as part FNP's response to the FNP 4B LAR (NRC RAI 08 Reference [34])).

Fire PRA Description	Fire PRA Sources of Uncertainty	Fire PRA Disposition
		The 50.69 categorization is a risk-ranking application that considers component functional importance. The methodology for the Cable Selection task does not introduce any epistemic uncertainties that would require sensitivity treatment.
Qualitative Screening	Qualitative screening was not performed; however, some structures (locations) were eliminated from the global analysis boundary and ignition sources deemed to have no impact on the FPRA (based on industry guidance and criteria) were excluded from the quantification based on qualitative screening criteria. The only criterion subject to uncertainty is the potential for plant trip. However, such locations would not contain any features (equipment or cables) identified in the prior two tasks and consequently are expected to have a low risk contribution.	In the event a structure (location) which could result in a plant trip was incorrectly excluded, its contribution to CDF would be small (with a CCDP commensurate with base risk). Such a location would have a negligible risk contribution to the overall FPRA. Based on the discussion of sources of uncertainty and the discussion above, it is concluded that the methodology for the Qualitative Screening task does not introduce any epistemic uncertainties that would require sensitivity treatment.
Fire-Induced Risk Model	A reactor trip is assumed as the initiating event for all quantification. The FPRA does not consider any special initiators (like loss of Service Water or Instrument Air) and does not consider turbine trip/MSIV closure events even though they may occur in a limited number of fire scenarios.	The identified source of uncertainty could result in the over-estimation of fire risk. In general, the FPRA development process has reviewed all significant fire initiating events and performed supplemental assessments to address this possible source of uncertainty. Based on the discussion of sources of uncertainty and the discussion above, it is concluded that the methodology for the Fire- Induced Risk Model task

Fire PRA Description	Fire PRA Sources of Uncertainty	Fire PRA Disposition
		does not introduce any epistemic uncertainties that would require sensitivity treatment.
Fire Ignition Frequencies	Fire ignition frequency is an area with inherent uncertainty. Part of this uncertainty arises due to the counting and related partitioning methodology. However, the resulting frequency is not particularly sensitive to changes in ignition source counts.	A consensus modeling approach is used to determine fire ignition frequencies. The primary source of uncertainty for this task is associated with the industry generic frequency values used for the FPRA. This is because there is no specific treatment for variability among plants along with some significant conservatism in defining the frequencies, and their associated heat release rates. Based on the discussion of sources of uncertainty, it is concluded that the methodology for the Fire Ignition Frequency task does not introduce any epistemic uncertainties that would require sensitivity treatment.
Quantitative Screening	Other than screening out potentially risk significant scenarios (ignition sources), this task is not a source of uncertainty.	The Farley FPRA development did not screen out any fire initiating events based on low CDF/LERF contribution. The methodology for the Quantitative Screening task does not introduce any epistemic uncertainties that would require sensitivity treatment.
Scoping Fire Modeling	The framework of NUREG/CR-6850 includes two tasks related to fire scenario development. These two tasks are Scoping Fire Modeling and Detailed Fire Modeling. The discussion of uncertainty for both tasks is provided in the discussion for Detailed Fire Modeling.	See Detailed Fire Modeling.

Fire PRA Description	Fire PRA Sources of Uncertainty	Fire PRA Disposition
Detailed Circuit Failure Analysis	The circuit analysis is performed using standard electrical engineering principles. However, the behavior of electrical insulation properties and the response of electrical circuits to fire induced failures is a potential source of uncertainty. This uncertainty is associated with the dynamics of fire and the inability to ascertain the relative timing of circuit failures. The analysis methodology assumes failures would occur in the worst possible configuration, or if multiple circuits are involved, at whatever relative timing is required to cause a bounding worst-case outcome. This results in a skewing of the risk estimates such that they are over-estimated.	Circuit analysis was performed as part of the deterministic post fire safe shutdown analysis. Refinements in the application of the circuit analysis results to the FPRA were performed on a case-by-case basis where the scenario risk quantification was large enough to warrant further detailed analysis. The uncertainty (conservatism) which may remain in the FPRA is associated with scenarios that do not contribute significantly to the overall fire risk. The methodology for the Detailed Circuit Failure Analysis task does not introduce any epistemic uncertainties that would require sensitivity treatment.
Circuit Failure Model Likelihood Analysis	One of the failure modes for a circuit (cable) given fire induced failure is a hot short. A conditional probability is assigned using industry guidance such as that published in NUREG/CR-6850. The uncertainty associated with the applied conditional failure probabilities poses competing considerations. On the one hand, a failure probability for spurious operation could be applied based solely on cable scope without consideration of less direct fire affects (e.g., a 0.3 failure likelihood applied to the spurious operation of a motor-operated valve (MOV) without consideration of the fire- induced	Uncertainty in the circuit failure mode likelihood analysis could lead to assumed failures of related components and related system functions. This would generate conservative results and that would typically be acceptable for most applications. Furthermore, a consensus modeling approach is used for Circuit Failure Mode Likelihood Analysis. Circuit failure mode likelihood analysis was generally limited to those components where spurious operation could not be caused by the generation of a spurious signal. This approach limited the introduction of non-conservative uncertainties.

Fire PRA Description	Fire PRA Sources of Uncertainty	Fire PRA Disposition
	generation of spurious signal to close or open the MOV). The analysis has biased the treatment such that it is assumed the spurious signal will always drive the valve in the unsafe direction. In addition, for those valves that might have multiple desired functions – consideration of spurious closure and consideration of failure to open on demand, the non-spurious failure state is treated with a logical TRUE rather than the complement of the spurious probability. For those valves that only have an active function, the potential for a spurious signal to drive the valve in the desired direction is ignored. The treatment results in skewing of the results such that the resulting risk is over-estimated.	For the 'simple' cases, the potential exists for assuming a failure likelihood greater than 0 in some areas where the cables capable of causing spurious operation are not located. Additional refinement to this approach was performed, as necessary, on risk significant scenarios. So the application of circuit failure probabilities is considered to have minimal impact on the results. The methodology for the Circuit Failure Mode Likelihood Analysis task does not introduce any epistemic uncertainties that would require sensitivity treatment.
Detailed Fire Modeling	The application of fire modeling technology is used in the FPRA to translate a fire initiating event into a set of consequences (fire induced failures). The performance of the analysis requires a number of key input parameters. These input parameters include the heat release rate (HRR) for the fire, the growth rate, the damage threshold for the targets, and the response of plant staff (detection, fire control, and fire suppression). The fire modeling methodology itself is largely empirical in some	A consensus modeling approach is used for Detailed Fire Modeling. Detailed fire modeling was performed only on those scenarios which otherwise would have been notable risk contributors and only where removal of conservatism in the generic fire modeling solution was likely to provide benefit either via a smaller zone of influence or to credit automatic suppression. The methodology for the Detailed Fire Modeling task does not introduce any epistemic uncertainties that would require sensitivity treatment.

Fire PRA Description	Fire PRA Sources of Uncertainty	Fire PRA Disposition
	respects and consequently is another source of uncertainty. For a given set of input parameters, the fire modeling results (temperatures as a function of distance from the fire) are characterized as having some distribution (aleatory uncertainty). The epistemic uncertainty arises from the selection of the input parameters (specifically the HRR and growth rate) and how the parameters are related to the fire initiating event. While industry guidance is available, that guidance is derived from laboratory tests and may not necessarily be representative of randomly occurring events. The fire modeling results using these input parameters are used to identify a zone of influence (ZOI) for the fire and cables/equipment within that ZOI are assumed to be damaged. In general, the guidance provided for the treatment of fires is conservative and the application of that guidance retains that conservatism. The resulting risk estimates are also conservative.	
Post-Fire Human Reliability Analysis	There are relatively few HFEs of high importance in the FPRA model. Conservative human error probability (HEP) adjustments were made to the nominal HEP values used in the FPIE model then revisited to address unique fire considerations. Given the	The human error probabilities were calculated using the EPRI HRAC and included the consideration of loss of necessary cues due to fire. The impact of any remaining uncertainties is expected to be small.

Fire PRA Description	Fire PRA Sources of Uncertainty	Fire PRA Disposition
	methodology used, the impact of any remaining uncertainties is expected to be small.	The methodology for the Post-Fire Human Reliability Analysis task does not introduce any epistemic uncertainties that would require sensitivity treatment.
Seismic-Fire Interactions Assessment	Since this is a qualitative evaluation, there is no quantitative impact with respect to the uncertainty of this task.	The qualitative assessment of seismic induced fires should not be a source of model uncertainties as it is not expected to provide changes to the quantified FPRA model. The methodology for the Seismic- Fire Interactions Assessment task does not introduce any epistemic uncertainties that would require sensitivity treatment.
Fire Risk Quantification	As the culmination of other tasks, most of the uncertainty associated with quantification has already been addressed. The other source of uncertainty is the selection of the truncation limit. However, the selected truncation limit is several orders of magnitude below the typical CDF value calculated and is consistent with the requirements of the PRA Standard.	The selected truncation was confirmed to be consistent with the requirements of the PRA Standard. The methodology for the Fire Risk Quantification task does not introduce any epistemic uncertainties that would require sensitivity treatment.
Uncertainty and Sensitivity Analyses	This task does not introduce any new uncertainties. This task is intended to address how the fire risk assessment could be impacted by the various sources of uncertainty.	This task does not introduce any new uncertainties. This task is intended to address how the fire risk assessment could be impacted by the various sources of uncertainty. The methodology for the Uncertainty and Sensitivity Analyses task does not introduce any epistemic uncertainties

Fire PRA Description	Fire PRA Sources of Uncertainty	Fire PRA Disposition
		that would require sensitivity treatment.
Fire PRA Documentation	FPRA Documentation This task does not introduce any new uncertainties to the fire risk.	This task does not introduce any new uncertainties to the fire risk as it outlines documentation requirements. The methodology for the FPRA documentation task does not introduce any epistemic uncertainties that would require sensitivity treatment.