Exelon Generation®

4300 Winfield Road
Warrenville, IL 60555
www.exeloncorp.com

10 CFR 50.90
10 CFR 50.69

RS-20-011

January 31, 2020

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

LaSalle County Station, Units 1 and 2
Renewed Facility Operating License Nos. NPF-11 and NPF-18
NRC Docket Nos. 50-373 and 50-374

Subject: 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, Exelon Generation Company, LLC (EGC) is requesting an amendment to the license to Renewed Facility Operating License Nos. NPF-11 and NPF-18 for LaSalle County Station, Units 1 and 2.

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

The PRA models described within this license amendment request (LAR) are the same as those described within the EGC submittal of the LAR dated January 31, 2020 for Risk-Informed Technical Specifications Initiative 4b, Risk-Managed Technical Specifications (RMTS) (RS-20-009). EGC requests that the NRC conduct their review of the PRA technical adequacy details for this application in coordination with the review of the application currently in-process. This would reduce the number of EGC and NRC resources necessary to complete the review of the applications. This request should not be considered a linked requested licensing action, as the

details of the PRA models in each LAR are complete which will allow the NRC staff to independently review and approve each LAR on their own merits without regard to the results from the review of the other.

EGC requests approval of the proposed license amendment by January 31, 2021, with the amendment being implemented within 60 days.

These proposed changes have been reviewed and approved by the LSCS Plant Operations Review Committee in accordance with the requirements of the EGC Quality Assurance Program.

In accordance with 10 CFR 50.91, "Notice for public comment; State consultation," paragraph (a)(1), the analysis about the issue of no significant hazards consideration using the standards in 10 CFR 50.92 is being provided to the Commission.

In accordance with 10 CFR 50.91, "Notice for public comment; State consultation," paragraph (b), EGC is notifying the State of Illinois of this application for license amendment by transmitting a copy of this letter and its attachments to the designated State Official.

There are no regulatory commitments contained within this letter.

Should you have any questions concerning this letter, please contact Ryan Sprengel at (630) 657-2814.

I declare under penalty of perjury that the foregoing is true and correct. Executed on the 31st day of January 2020.

Respectfully,

Dwi Murray
Sr. Manager – Licensing
Exelon Generation Company, LLC

Enclosure: Evaluation of the Proposed Change

cc: NRC Regional Administrator, Region III
NRC Senior Resident Inspector – LaSalle County Station
Illinois Emergency Management Agency – Division of Nuclear Safety
NRR Project Manager, LaSalle County Station

Enclosure

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Evaluation of the Proposed Change

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 Exelon Generation Company, LLC (EGC) to improve focus on equipment that has safety significance resulting in improved plant safety.

2.3 DESCRIPTION OF THE PROPOSED CHANGE

EGC proposes the addition of the following condition to the renewed operating license of LSCS, Units 1 and 2, to document the NRC's approval of the use 10 CFR 50.69.

> Exelon Generation Company, LLC (EGC) 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 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 the EGC submittal letter dated January 31, 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.

The PRA models described within this LAR are the same as those described within the EGC submittal of the LAR dated January 31, 2020 for Risk-Informed Technical Specifications Initiative 4b, Risk-Managed Technical Specifications (RMTS) (RS-20-009). EGC requests that the NRC conduct their review of the PRA technical adequacy details for this application in coordination with the review of the application currently in-process. This would reduce the number of EGC and NRC resources necessary to complete the review of the applications. This request should not be considered a linked requested licensing action (RLA), as the details of the PRA models in each LAR are complete which will allow the NRC staff to independently review and approve each LAR on their own merits without regard to the results from the review of the other.

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

3.1.1 Overall Categorization Process

EGC 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.

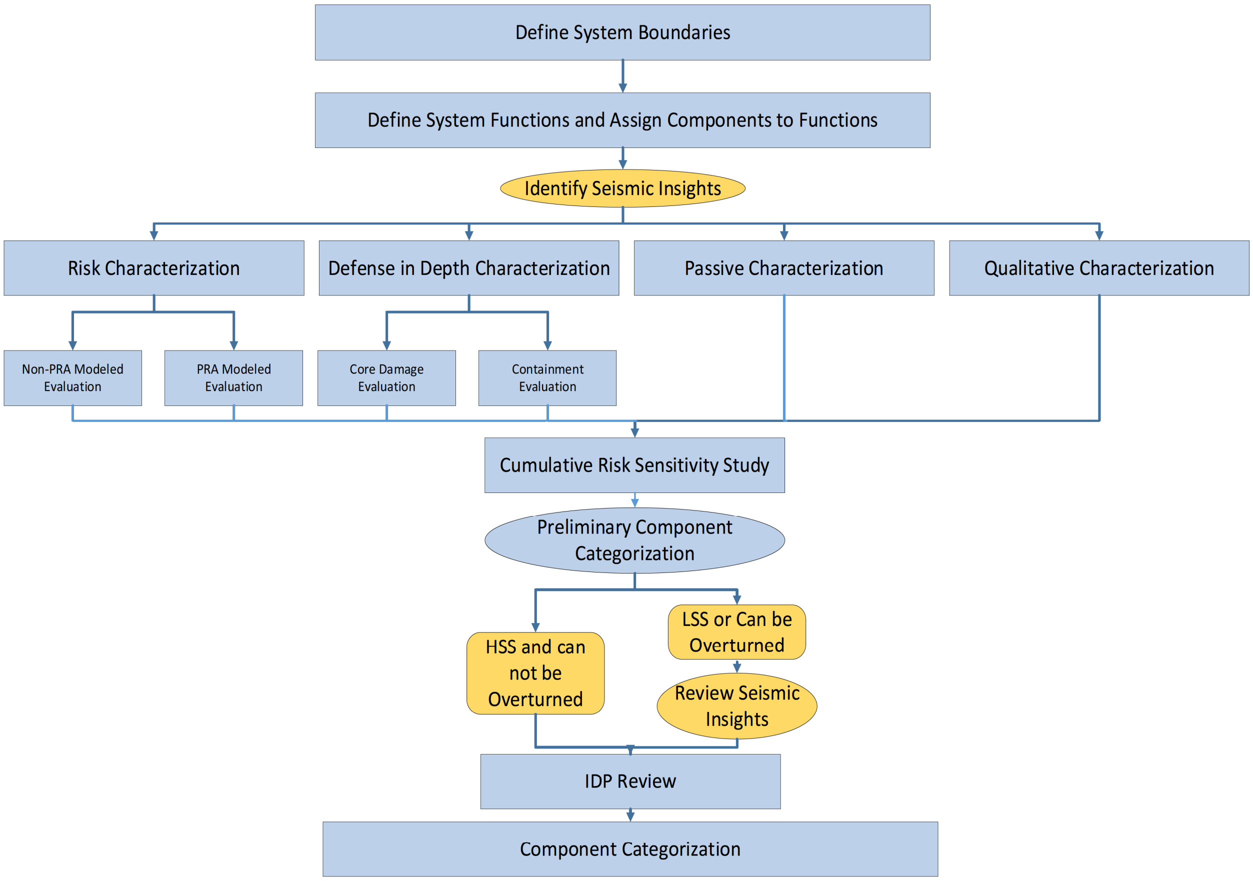
The process to categorize each system will be consistent with the guidance in NEI 00-04, "10 CFR 50.69 SSC Categorization Guideline," as endorsed by RG 1.201, with the exception of the evaluation of impact of the seismic hazard, which will use the EPRI 3002012988 [3][1] approach for seismic Tier 2 sites, which includes LSCS, 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:

[1] Updates to EPRI 3002012988 report [3] are incorporated by reference into this LSCS submittal. These updates are cited in Attachment 2 of the EGC RAI response dated July 19, 2019 for Calvert Cliffs' 10 CFR 50.69 LAR (ML19200A216) [60].

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

<u>Notes</u>:
[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 LSCS is a seismic Tier 2 (moderate 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 EGC 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, EGC will not take credit for alternate means unless the alternate means are proceduralized and included in Licensed Operator training.

- LSCS 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 3002012988 (Reference [3]) for Tier 2 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, as submitted to the NRC for TSTF-505 dated January 31, 2020 (RS-20-009) (Refer to Attachment 2).

- Fire Risks: Fire PRA model, as submitted to the NRC for TSTF-505 dated January 31, 2020 (RS-20-009) (Refer to Attachment 2).

- Seismic Risks: EPRI Alternative Approach in EPRI 3002012988 (Reference [3]) for Tier 2 plants with the additional considerations discussed in Section 3.2.3 of this LAR.

- Other External Risks (e.g., tornados, external floods): Using the IPEEE screening process as approved by NRC SE dated December 8, 2000 (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" (Reference [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., defense in depth, 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 LSCS 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. The PRA models credited in this request are the same PRA models credited in the TSTF-505 application dated January 31, 2020, (RS-20-009).

3.2.1 Internal Events and Internal Flooding

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

3.2.2 Fire Hazards

The LSCS 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 EGC risk management process ensures that the PRA model used in this application reflects the as-built and as-operated plant for LSCS. 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 in the absence of a quantifiable PRA (such as Seismic Margin Analysis or IPEEE Screening) as part of an integrated, systematic process. For the LSCS seismic hazard assessment, EGC Nuclear 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 Reference [3] and includes additional considerations that are discussed in this section.

The proposed categorization approach for LSCS is a risk-informed graded approach that is demonstrated to produce categorization insights equivalent to a seismic PRA. This approach relies on the insights gained from the seismic PRAs examined in Reference [3] and plant specific insights considering seismic correlation effects and seismic interactions. Following the criteria in Reference [3], the LSCS site is considered a Tier 2 site because the site GMRS to SSE comparison is above the Tier 1 threshold but not high enough that the NRC required the plant to perform an SPRA to respond to Recommendation 2.1 of the Near Term Task Force 50.54(f) letter (Reference [8]). Reference [3] also demonstrates that seismic risk is adequately addressed for Tier 2 sites by the results of additional qualitative assessments discussed in this section and existing elements of the 50.69 categorization process specified in NEI 00-04.

For example, the 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 results of the integral assessment meets the importance measure criteria for LSS. In applying the EPRI 3002012988 (Reference [3]) process to the 10 CFR 50.69 categorization process, the Integrated Decision-making Panel (IDP) will be provided with the rationale for applying the EPRI 3002012988 guidance and informed of plant SSC-specific seismic insights for their consideration in the HSS/LSS deliberations.

The trial studies in Reference [3], as amended by their RAI responses and amendments (References [9], [10], [11], [12], and [13]), 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 2 classification and resulting criteria 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 Reference [3].

> *"At Tier 2 sites, there may be a limited number of unique seismic insights, most likely attributed to the possibility of seismically correlated failures, appropriate for consideration in determining HSS SSCs. The special seismic risk evaluation process recommended using a Common Cause impact approach in the FPIE PRA can identify the appropriate seismic insights to be considered with the other categorization insights by the Integrated Decision-making Panel for the final HSS determinations."*

At sites with moderate seismic demands (i.e., Tier 2 range) such as LSCS, there is no need to perform more detailed evaluations to demonstrate the inherent seismic capacities documented in industry sources such as Reference [14]. Tier 2 seismic demand sites have a lower likelihood of seismically induced failures and less challenges to plant systems. This, therefore, provides the technical basis for allowing use of a graded approach for addressing seismic hazards at LSCS.

Test cases described in Section 3 of Reference [3], as amended by their RAI responses and amendments (References [9], [10], [11], [12], and [13]), showed there are very few, if any, SSCs that would be designated HSS for seismic unique reasons. The test cases identified that the unique seismic insights were typically associated with seismically correlated failures and led to unique HSS SSCs. While it would be unusual even for moderate hazard plants to exhibit any unique seismic insights, it is prudent and recommended by Reference [3] to perform additional evaluations to identify the conditions where correlated failures and seismic interactions may occur and determine their impact in the 50.69 categorization process. The special sensitivity study recommended in Reference [3] uses common cause failures, similar to the approach taken in a FPIE PRA and can identify the appropriate seismic insights to be considered with the other categorization insights by the IDP for the final HSS determinations.

EGC is using test case information from Reference [3], developed by other licensees. The test case information is being incorporated by reference into this application, specifically Case Study A (Reference [15]), Case Study C (Reference [16]), and Case Study D (Reference [17]) as well as, RAI responses and amendments (References [9], [10], [11], [12], and [13]), clarifying aspects these case studies.

<u>Basis for LSCS being a Tier 2 Plant</u>

As defined in Reference [3], LSCS meets the Tier 2 criteria for a "Moderate Seismic Hazard / Moderate Seismic Margin" site. The Tier 2 criteria are as follows:

> *"Tier 2: Plants where the GMRS [Ground Motion Response Spectrum] to SSE [Safe Shutdown Earthquake] comparison between 1.0 Hz and 10 Hz is greater than in Tier 1 but not high enough to be treated as Tier 3. At these sites, the unique seismic categorization insights are expected to be limited."*

Note: Reference [3] applies to the Tier 2 sites in its entirety except for Sections 2.2 (Tier 1 sites) and 2.4 (Tier 3 sites).

For comparison, Tier 1 plants are defined as having a GMRS peak acceleration at or below approximately 0.2g or where the GMRS is below or approximately equal to the SSE between 1.0Hz and 10 Hz. Tier 3 plants are defined where the GMRS to SSE comparison between 1.0 Hz and 10 Hz is high enough that the NRC required the plant to perform an SPRA to respond to the Fukushima 50.54(f) letter (Reference [8]).

As shown in Figure 2 in Section 2.5, comparing the LSCS GMRS (derived from the seismic hazard) to the SSE (i.e. seismic design basis capability), the GMRS is largely below the SSE up through 6 Hz and exceeds the SSE above 6 Hz. (Reference [18]). The NRC screened out LSCS from performing an SPRA in response to the NTTF 2.1 50.54(f) letter (Reference [19]). As such, it is appropriate that LSCS is considered a Tier 2 plant. The basis for LSCS being Tier 2 will be documented and presented to the IDP for each system categorized.

The following paragraphs describe additional background and the process to be utilized for the graded approach to categorize the seismic hazard for a Tier 2 plant.

<u>Implementation of the Recommended Process</u>

Reference [3] recommends a risk-informed graded approach for addressing the seismic hazard in the 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 report. The coupling of these concepts with the categorization process in NEI 00-04 are the key elements of the approach defined in Reference [3] 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 [14]) 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 and that same seismic capacity applies at a site with a very low demand and at a site with a very high demand.

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.

In applying the Reference [3] process for Tier 2 sites to the LSCS 10 CFR 50.69 categorization process, the IDP will be provided with the rationale for applying the Reference [3] guidance and informed of plant SSC-specific seismic insights that the IDP may choose to consider in their HSS/LSS deliberations. 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 that provides identified plant seismic insights as well as the basis for applicability of the Reference [3] study and the bases for LSCS being a Tier 2 plant. The discussion of the Tier 2 bases will include such factors as:

- The moderate seismic hazard for the plant,
- The definition of Tier 2 in the EPRI study, and
- The basis for concluding LSCS is a Tier 2 plant.

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 LSCS) 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 LSCS 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 LSCS plant-specific seismic reviews and other resources such as those identified above. The objective of the seismic review is to identify plant-specific seismic insights that might include potentially important impacts such as:

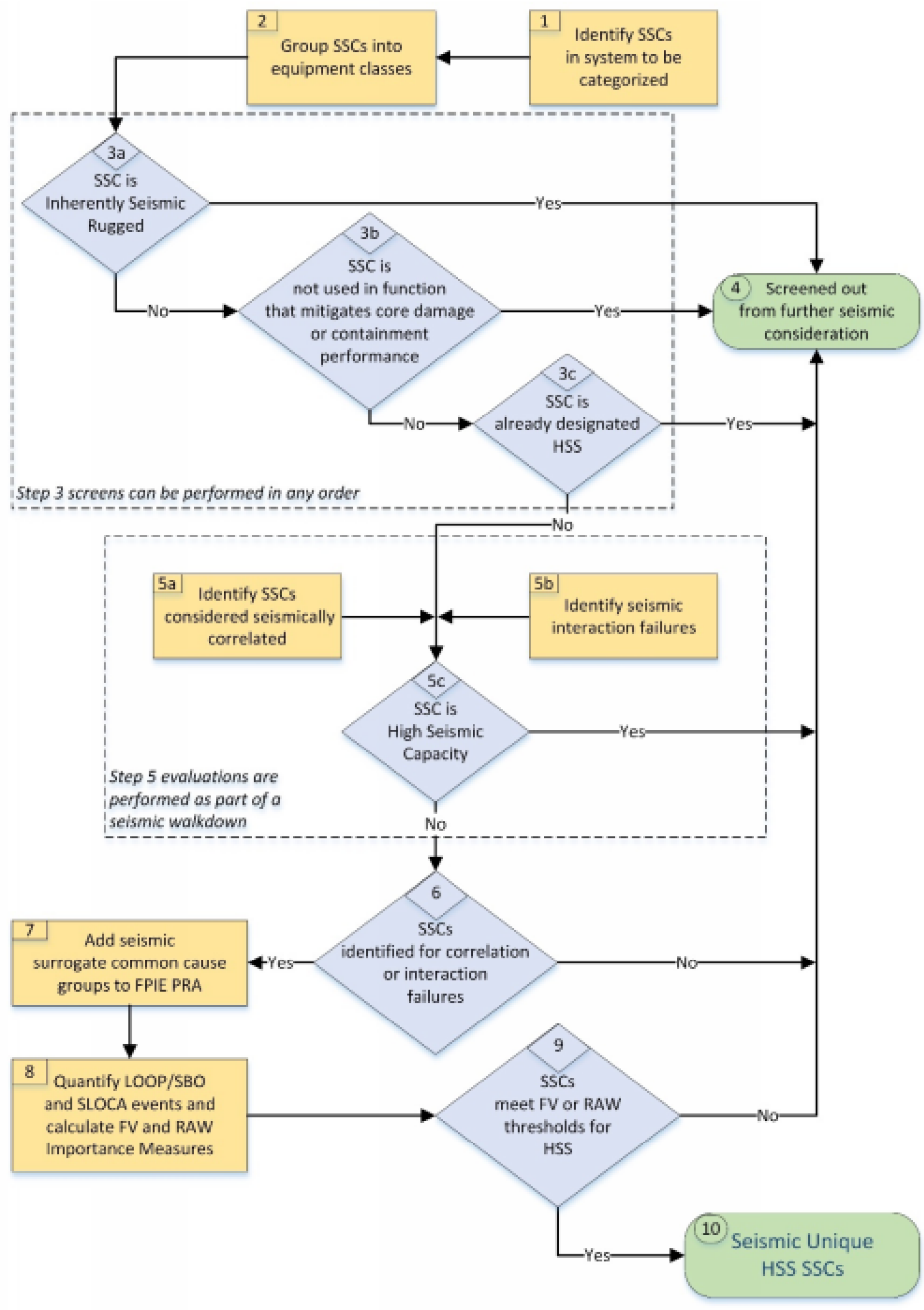
- Impact of relay chatter
- Implications related to potential seismic interactions such as with block walls
- Seismic failures of passive SSCs such as tanks and heat exchangers
- Any known structural or anchorage issues with a particular SSC
- Components implicitly part of PRA-modeled functions (including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

For each system categorized, the categorization team will evaluate correlated seismic failures and seismic interactions between SSCs. This process is detailed in Reference [3] Section 2.3.1 and is summarized below in Figure 3-2. Determination of seismic insights will make use of the full power internal events PRA model supplemented by focused seismic

walkdowns. To determine the importance of SSCs for mitigating seismic events the following process will be utilized on a system basis:

- Identify SSCs within the system to be categorized
- Group SSCs into equipment classes according to Reference [14] and separate-out distribution systems such as cable tray, piping, and HVAC ductwork.
- Refine the list and screen:
  - Inherently rugged components like check valves, manual valves, and valves (AOV and MOV) not required to change state.
  - Screen if the component is not used in safety functions that support mitigation of core damage or containment performance
  - Screen if the component is already identified as a HSS component from the Internal Events PRA or Integrated assessment
  - Do a seismic walkdown on SSCs screened IN to look for correlation and spatial interaction concerns
  - Based on the seismic walkdown, screen out IF SSCs have high seismic capacity AND not included in seismically correlated groups or correlated interaction groups
- Add surrogate events to the FPIE model that simulate spatial interaction or correlation- set the probability of failure to 1E-04 or justify based on the hazard
- Quantify the FPIE model for LOOP and Small LOCA (SLOCA) accident sequences setting the LOOP initiating event frequency to 1.0/yr and the SLOCA initiating event frequency to 1E-02/yr
- Utilize the Importance Measures from this sensitivity study to identify appropriate SSCs that should be HSS due to correlation or seismic interactions

Figure 3-2: Seismic Correlated Failure Assessment for Tier 2 Plants[2]



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[2] Reproduced from Reference [3] Figure 2-3

Such impacts would be compiled on an SSC basis. As each system is categorized, the system-specific seismic insights will be documented in the categorization report and provided to the IDP for consideration as part of the IDP review process (e.g., Figure 3-1). The IDP can challenge any candidate HSS recommendation for any SSC from a seismic perspective if they believe there is a basis. Any decision by the IDP to downgrade preliminary HSS components to LSS will consider the applicable seismic insights in that decision. SSCs identified from the Fire PRA as candidate HSS, which are not HSS from the internal events PRA or integrated importance measure assessment, will be reviewed for their design basis function during seismic events or functions credited for mitigation and prevention of severe accidents caused by seismic events. These insights will provide the IDP a means to consider potential impacts of seismic events in the categorization process.

In the unlikely event that the LSCS seismic hazard changes from medium risk (i.e., Tier 2) at some future time, EGC 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).

<u>Historical Seismic References for LSCS County Generating Station</u>

The LSCS SSE and GMRS curves from the seismic hazard and screening response are shown in Section 2.4 and 3.1 of Reference [20]. The LSCS SSE and GMRS curves from the seismic hazard and screening response are shown in Figure A4-1 of Attachment 4. The NRC's Staff assessment of the LSCS seismic hazard and screening response is documented in Reference [19]. In the Staff Confirmatory Analysis (Section 3.3.3) on page 10 of Reference [19], the NRC concluded that the methodology used by EGC in determining the GMRS was acceptable and that the GMRS determined by EGC adequately characterizes the reevaluated hazard for the LSCS 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 LSCS, 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 [19] and [20]).
2. NTTF Recommendation 2.1 spent fuel pool assessment (References [21] and [22]).
3. NTTF Recommendation 2.3 seismic walkdowns (References [23] and [24]).
4. NTTF Recommendation 4.2 seismic mitigation strategy assessment (S-MSA) (References [25] and [26]).

The following additional post-Fukushima seismic reviews were performed for LSCS:

5. NTTF Recommendation 2.1 seismic Expedited Seismic Evaluation Process (ESEP) (References [27], [28])
6. NTTF Recommendation 2.1 seismic High Frequency Evaluation (References [29] and [30]

Summary

Based on the above, the Summary from Section 2.3.3 of Reference [3] applies to LSCS; namely, LSCS is a Tier 2 plant for which there may be a limited number of unique seismic insights, most likely attributed to the possibility of seismically correlated failures, appropriate for consideration in determining HSS SSCs. The special sensitivity study recommended using common cause failures, similar to the approach taken in a FPIE PRA, can identify the appropriate seismic insights to be considered with the other categorization insights by the Integrated Decision-making Panel (IDP) for the final HSS determinations. Use of the EPRI approach outlined in Reference [3] to assess seismic hazard risk for 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).

3.2.4 Other External Hazards

All external hazards, except for seismic, were screened for applicability to LSCS per a plant-specific evaluation in accordance with GL 88-20 (Reference [31]) 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 LSCS 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 EGC risk management process ensures that the applicable PRA models used in this application continues to reflect the as-built and as-operated plant for LSCS. 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, EGC 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 of NEI 00-04.

In the overall risk sensitivity studies, EGC 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, EGC 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 [32]). 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 LSCS 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 LSCS 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 LSCS 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 have 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 [33]), consistent with NRC RIS 2007-06.

The Internal Events PRA model received a formal industry peer review in April 2008. The FPIE Peer Review was performed using the NEI 05-04 process, the ASME PRA Standard ASME RA-Sc-2007, and Regulatory Guide 1.200, Rev. 1. The 2008 Internal Events PRA Peer Review findings were addressed in subsequent PRA updates and a F&O Closure Review was performed by an independent review team in June 2017 (Reference [34]).

Eleven (11) F&Os associated with SRs assessed as less than Capability Category II (i.e., SRs assessed as "Not Met" or Capability Category I) were categorized as suggestions rather than findings. The resolution of these suggestion level F&Os was not previously independently reviewed, so a supplemental FPIE F&O Closure Review was performed in conjunction with the Fire PRA closure in October 2017 (Reference [35]).

Since the peer review of the Internal Events PRA model was performed prior to the publication of RG 1.200 Rev. 2, a self-assessment was conducted to assess the differences between RE 1.200 Rev. 2 and RG 1.200 Rev. 1 [36]. That assessment confirmed that the PRA model meets the requirements of RG 1.200 Rev. 2 and the results from that assessment are documented in Attachment 7.

The Fire PRA model received a formal industry peer review in December 2015. The Fire PRA Peer Review was performed using the NEI 07-12 Fire PRA peer review process, the ASME/ANS PRA Standard, ASME/ANS RA-Sa-2009, and Regulatory Guide 1.200, Rev. 2. The 2015 LSCS Fire PRA Peer Review was a full-scope review of the LSCS at-power Fire PRA against all technical elements in Part 4 of the ASME/ANS PRA Standard, including the referenced Internal Events Supporting Requirements (SRs).

The 2015 Fire PRA Peer Review findings were addressed in subsequent PRA updates and a F&O Closure Review was performed by an independent review team in October 2017 (Reference [35]). During the October 2017 F&O Closure Review, a Focused Scope Peer Review (FSPR) was conducted against the Fire Risk Quantification (FQ) Technical Element due to the large reduction in CDF and LERF as a result of the resolution to several technical F&Os and other model refinements.

In September 2019, another F&O Closure Review (Reference [37]) was conducted to independently review the remaining open F&Os (including the new F&Os from the FSPR).

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 [38]) as accepted by NRC in the letter dated May 3, 2017 (ML17079A427) (Reference [39]). The results of this review have been documented and are available for NRC audit.

Attachment 3 provides a summary of the remaining findings and open items, including:

- Open items and disposition from the LSCS RG 1.200 self-assessment.

- Open findings and disposition of the LSCS peer reviews.

The attachments identified above demonstrate 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 10 CFR 50.69(c)(1)(i).

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

The LSCS 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 LSCS Tier 2 approach discussed in section 3.2.3, implementation of the EGC 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 the EGC 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 help catch and reverse negative performance trends and take corrective action if necessary.

The EGC 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 impact to seismic loading, safe shutdown earthquake (SSE) seismic requirements, as well as the method of combining seismic components.
- Review of seismic dynamic qualification of components if the configuration change adds, relocates, or alters Seismic Category I mechanical or electrical components.

EGC 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 EGC 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 no longer than once every two refueling outages 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 updated, 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

EGC 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.

EGC 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, EGC 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.

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Attachment 1: List of Categorization Prerequisites

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

- 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

Plant	Units	Model	Baseline CDF	Baseline LERF	Comments
LaSalle	1 & 2	**Full Power Internal Events and Internal Flooding PRA** LS216C (Unit 1&2)	1.3E-06 (Unit 1) 1.3E-06 (Unit 2)	1.3E-07 (Unit 1) 1.3E-07 (Unit 2)	Application Specific Model (ASM) to the Full Power Internal Events (FPIE) with Internal Flooding update which was based on the 2014 PRA MORs
	1 & 2	**Fire PRA** LS114AF3 (Unit 1) LS214AF3 (Unit 2)	1.0E-05 (Unit 1) 7.8E-06 (Unit 2)	9.8E-07 (Unit 1) 3.2E-07 (Unit 2)	Application Specific Model (ASM) to the Fire Update which was based on the 2014 PRA MORs

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
Internal Events F&Os				
IE-D3-01 (Finding)	IE-D3 AS-C3 SC-C3 CY-C3 HR-I3 DA-D3 IF-F3 QU-E2 LE-G4	Not Met CCII	The Summary Notebook includes information that attempts to identify the key sources of uncertainty in the initiating event analysis. However, with the changes to eliminate "key" from the SR definition, this SR cannot be considered met. Section 4 of the LS-PSA-013 notebook (Reference [40]) discusses the industry "key sources of uncertainty" per EPRI guidance. However, the current analysis does not fully meet the requirements of RG 1.200, which requires a discussion of sources of model uncertainty and related assumptions. Also, there may be some plant-specific assumptions made that may not be fully captured by the generic list of potential sources of uncertainty.	Additional documentation of LERF key sources of uncertainty including results and important insights are needed to fully close out this Finding. However, this issue does not impact 50.69 applications. The model sources of uncertainty, both generic and plant-specific, as they impact this risk-informed application are specifically addressed in Attachment 6 of this LAR.

Finding Number	Supporting Requirement(s)	Capability Category (CC)	Description	Disposition for 50.69
SY-A4-01 (Suggestion)	SY-A4	Not Met CCII	Enhance PRA technical capability. Perform plant walkdowns with system engineers AND plant operators. Better document the walkdowns performed in support of the PRA and reference those walkdowns in each system notebook to achieve Capability Category II	While it is judged that this Finding has no impact on the PRA results and therefore, no impact on 50.69 implementation, this F&O will be resolved during a LSCS PRA update and system walkdowns will be conducted and documented with System Engineers and Plant Operators prior to implementation of 50.69.

Finding Number	Supporting Requirement(s)	Capability Category (CC)	Description	Disposition for 50.69
DA-C8-01 (Finding)	DA-C8	Not Met CCII	Basic events used to model the standby status of various plant systems use a mix of plant-specific operational data and engineering judgment. For the plant service water system and several other systems, standby estimates have been determined from procedures and operating data. For other components, assumptions are used (e.g., 50% probability of either of two pumps in a system is in standby). So, overall the LSCS PRA has some Capability Category (CC) II attributes and some CC-I attributes. Current approach of assuming standby time does not meet the requirements of the Supporting Requirement. The use of actual plant data could result in small changes in PRA results.	While it is judged that this Finding has no impact on the PRA results and therefore, no impact on 50.69 implementation, this F&O will be resolved during a LSCS PRA update and plant-specific data reviews will be performed and documented prior to implementation of 50.69.

Finding Number	Supporting Requirement(s)	Capability Category (CC)	Description	Disposition for 50.69
DA-C6-01 (Suggestion)	DA-C7 DA-C9 DA-C10	Not Met CCII	LS-PSA-010, Component Data Notebook, Appendix C, states, "No actual data or estimates for these parameters are provided by system managers. Data from the MSPI basis document, scoping and performance criteria document, and 2003 data notebook is used." As the was obtained from Maintenance Rule and MSPI sources, the techniques used to obtain this data are probably consistent with the guidance in this supporting requirement, but this cannot be positively determined. Similarly, for SR DA-C7, it is unable to be determined if surveillance tests, planned and unplanned maintenance activities were based on actual plant experience. For SR DA-C9, the reviewers were unable to conclude whether plant specific operational records were used to determine standby time. Similarly, for DA-C10, it is not clear how surveillance tests were used. This appears to be primarily a documentation issue, as it is expected that the assumptions used to collect data for Maintenance Rule and MSPI are similar to those required by the ASME standard. However, it is possible that some differences in methodology could exist between these programs and the PRA.	This issue has minimal impact on the 50.69 application as the plant-specific data was updated during the 2011 and 2014 PRA updates. Further, LSCS will be updating the plant-specific data during a PRA update before implementation of 50.69.

Finding Number	Supporting Requirement(s)	Capability Category (CC)	Description	Disposition for 50.69
IF-C3b-01 (Suggestion)	IF-C3B	Not Met CCII	Address potential unavailability of barriers that affect the propagation of water in order to meet the CC II requirements of the ASME Standard. This is a suggestion since it is considered a documentation issue. The flood scenarios analyzed in detail are so large (i.e., typically involving draining the lake into the Turbine building until it fills) that structural analysis of non-flood doors and any difference in flood propagation will have no significant impact.	This open issue has no impact on the 50.69 implementation as it is a documentation issue. However, this suggestion will be resolved during a PRA update.

Finding Number	Supporting Requirement(s)	Capability Category (CC)	Description	Disposition for 50.69
Fire Model F&Os				
1-19 (Suggestion)	CS-A1	NOT MET CCII	The peer review examined the cable selection package for offsite power loss switchyard breaker (OCB 4-6). The circuit evaluation package includes two pages of notes regarding interlock evaluations and the notes and assumptions associated with the interlocks. For example, a note is made that "the interlock associated with trip and lockout of SAT 242. Cables that can cause relay to actuate are to be included with SAT 242". The FPRA development team indicated that this impact for SAT 242 is addressed by the FPRA, but that no systematic review of the circuit evaluation package notes was performed. A review of circuit evaluation notes and assumptions is important to ensure that FPRA plant response model identifies cables whose fire-induced failure could adversely affect selected equipment and/or credited functions in the Fire PRA plant response model.	This item has no impact on the 50.69 as it has been resolved, just not reviewed and closed by the Independent Assessment Team.

Finding Number	Supporting Requirement(s)	Capability Category (CC)	Description	Disposition for 50.69
4-17 (Finding)	FSS-D7	Not Met CCII	There is no generic estimate or plant-specific value assigned to the non-suppression probability. The non-suppression values are only based on the NUREG/CR-6850 generic values for unreliability with no account for unavailability.	The impact on the 50.69 LAR is judged to be minimal. However, plant-specific data will be reviewed and refined data for automatic detection and suppression systems will be incorporated into the FPRA model during a Fire PRA update if necessary. This item will be resolved prior to 50.69 implementation.
6-11 (Finding)	CS-A1 CS-A2 CS-A3	Not Met CCII	The cable selection work performed related to the cable data in the fire safe shutdown report pre-dates NEI-00-01 and was done to the standards at that time. No other information is currently available regarding the circuit analysis techniques used for the fire safe shutdown report. In general, the MSO circuit analysis work was performed using NEI-00-01, Revision 2 or Revision 3 (depending upon the particular package).	There is no impact on 50.69 implementation as this issue will be resolved prior to implementation.

Attachment 4: External Hazards Screening

External Hazard	Screening Result		
	Screened? (Y/N)	**Screening Criterion (Note a)**	**Comment**
Aircraft Impact	Y	PS2 PS4	In the NRC Staff Evaluation of the IPEEE (Reference [5]), a probabilistic bounding analysis was performed for aircraft impact. The median frequency of CDF was calculated as 5E-7/year **(PS4)**. **(PS2)**: From Section 3.5.1.6, Aircraft Hazards of the LSCS UFSAR, the airports and airways in the vicinity of the site are described in Subsection 2.2.2.5 of the UFSAR (Reference [41]). a. There are no federal airways or airport approaches passing within 2 miles of the station. The closest airway corridor is 3 miles away from the station. b. There are no commercial airports existing within 10 miles of the site and there is only one private airstrip within 5 miles. c. The projected landing and take-off operations out of those airports located within 10 miles of the site are far less than 500 d^2 per year, where d is the distance in miles. The projected operations per year for airports located outside of 10 miles is less than 1000 d^2 per year. d. The only military facility within 10 miles of the site is the Illinois Army Reserve National Guard Training Facility. It is located approximately 1 mile northwest of LSCS cooling lake. There are no airstrips at the Training Facility.

External Hazard	**Screening Result**		
	Screened? (Y/N)	**Screening Criterion (Note a)**	**Comment**
			Based on this review, the Aircraft Impact hazard can be considered to be negligible.
Avalanche	Y	C3	The mid-western location of LSCS station precludes the possibility of an avalanche. Based on this review, the Avalanche impact hazard can be considered to be negligible.
Biological Event	Y	C5	Hazard is slow to develop and can be identified via monitoring and managed via standard maintenance process. Actions committed to and completed by LSCS station in response to Generic Letter 89-13 provide on-going control of biological hazards. These controls are described in EGC procedure ER-AA-340, "GL 89-13 Program Implementing Procedure" (Reference [42]). Based on this review, the Biological Event impact hazard can be considered to be negligible.
Coastal Erosion	Y	C3	The mid-western location of LSCS station precludes the possibility of coastal erosion. Based on this review, the Coastal Erosion impact hazard can be considered to be negligible.
Drought	Y	C5	Drought is a slowly developing hazard allowing time for orderly plant reductions, including shutdowns. Based on this review, the Drought impact hazard can be considered to be negligible.

External Hazard	Screening Result		
	Screened? (Y/N)	Screening Criterion (Note a)	Comment
External Flooding	Y	C1	The external flooding hazard at the site was recently updated as a result of the post-Fukushima 50.54(f) Request for Information. A flood hazard reevaluation report (FHRR) was submitted to NRC for review on March 12, 2014 [43]. The results indicate that flooding from all mechanisms except local intense precipitation (LIP) and probable maximum storm surge (PMSS) were bounded by the current licensing basis (CLB). Only LIP and PMSS require evaluation in a Focused Evaluation (FE) to determine if the plant's current design basis bounds the reevaluated flood parameters. Further investigation was performed and the results of the FE were submitted to NRC for review and a staff assessment was issued on August 23, 2017 [44]. The NRC acknowledged the results presented in the FE concluding that there were no impacts to SR SSCs from the LIP and PMSS events and the design basis of the plant is adequate to mitigate the effects from external flood causing mechanisms with sufficient margin. In accordance with the external hazard screening process per Figure 5-6 of NEI 00-04, several flood doors integral to flood protection at LSCS were identified for categorization as High Safety Significant (HSS) SSCs should their associated systems be categorized. Based on this review, the External Flooding impact hazard can be considered to be negligible.

External Hazard	**Screening Result**		
	Screened? (Y/N)	**Screening Criterion (Note a)**	**Comment**
Extreme Wind or Tornado	Y	C1	Based on the plant design for wind pressure and the low frequency (<1E-7/yr) of design tornadoes, a demonstrably conservative estimate of CDF associated with high wind hazard (other than wind generated missiles) is much less than 1E-6/yr. In addition, based on the plant design for tornado missiles, considering a limited set of SSCs vulnerable to tornado missiles, a demonstrably conservative estimate of CDF associated with tornado missiles is less than 1E-6/yr. Based on this review, the Extreme Wind or Tornado impact hazard can be considered to be negligible.
Fog	Y	C4	The principal effects of such events (such as freezing fog) would be to cause a loss of off-site power and are addressed in the weather-related Loss of Offsite Power initiating event in the internal events PRA model for LSCS. Based on this review, the Fog impact hazard can be considered to be negligible.
Forest or Range Fire	Y	C3	Forest fires were screened in the IPEEE (Reference [45]). The site landscaping and lack of forestation prevent such fires from posing a threat to LSCS station. Based on this review, the Forest or Range Fire impact hazard can be considered to be negligible.

External Hazard	Screening Result		
	Screened? (Y/N)	**Screening Criterion (Note a)**	**Comment**
Frost	Y	C4	The principal effects of such events would be to cause a loss of off-site power and are addressed in the weather-related Loss of Offsite Power initiating event in the internal events PRA model for LSCS. Based on this review, the Frost impact hazard can be considered to be negligible.
Hail	Y	C4	The principal effects of such events would be to cause a loss of off-site power and are addressed in the weather-related Loss of Offsite Power initiating event in the internal events PRA model for LSCS. Based on this review, the Hail impact hazard can be considered to be negligible.
High Summer Temperature	Y	C1 C4	The plant is designed for this hazard **(C1)**. The principal effects of such events would result in elevated lake temperatures which are monitored by station personnel. Should the ultimate heat sink temperature exceed the LSCS Technical Specification 3.7.3 temperature limit, an orderly shutdown would be initiated. In addition, plant trips due to this hazard are covered in the definition of another event in the PRA model (e.g., transients, loss of condenser) **(C4)**. Based on this review, the High Summer Temperature impact hazard can be considered to be negligible.
High Tide, Lake Level, or River Stage	Y	C3 C5	The mid-western location of LSCS station precludes the possibility of a high tide condition **(C3)**.

External Hazard	Screening Result		
	Screened? (Y/N)	**Screening Criterion (Note a)**	**Comment**
			High lake effects would take place slowly allowing time for orderly plant reductions including shutdowns **(C5)**. Based on this review, the High Tide, Lake Level, or River Stage impact hazard can be considered to be negligible.
Hurricane	Y	C3	The mid-western location of LSCS station precludes the possibility of a hurricane. Based on this review, the Hurricane impact hazard can be considered to be negligible.
Ice Cover	Y	C1 C4	Per UFSAR 2.4.7 (Reference [41]), essential for ice jam formation is a constriction to passage of flowing ice. Such a constriction does not exist in the Illinois River near the site, since the river is approximately 800 feet wide and is kept navigable by dredging when required. The lake screen house is protected against icing in the lake by provision of warming lines near the screen house **(C1)**. The principal effects of such events would be to cause a loss of off-site power and are addressed in the weather-related Loss of Offsite Power initiating events in the internal events PRA model for Lasalle **(C4)**. Based on this review, the Ice Cover impact hazard can be considered to be negligible.
Industrial or Military Facility Accident	Y	C1 C3	The only military facility within 10 miles is the Illinois Army Reserve National Guard (ILARNG) Training Facility within 1 mile northwest of LSCS Station and encompassing

External Hazard	Screening Result		
	Screened? (Y/N)	Screening Criterion (Note a)	Comment
			approximately 2560 acres. There are no missile sites, bombing ranges or runways at the facility, but there are 5 firing ranges in the direction of north to northwest (**C3**). Hazardous chemicals used and/or stored by manufacturers within five miles of the plant were also evaluated and determined to either screen from further evaluation or were determined to meet the acceptance criteria associated with Control Room operator protection as discussed in LSCS UFSAR, Section 2.2.3 (Reference [41]) (**C1**) Based on this review, the Industrial or Military Facility Accident impact hazard can be considered to be negligible.
Internal Flooding	N/A	N/A	The LSCS Internal Events PRA includes evaluation of risk from internal flooding events.
Internal Fire	N/A	N/A	The LSCS Internal Fire PRA includes evaluation of risk from internal fire events
Landslide	Y	C3	The mid-western location of LSCS station precludes the possibility of a landslide. Based on this review, the Landslide impact hazard can be considered to be negligible.

External Hazard	**Screening Result**		
	Screened? (Y/N)	**Screening Criterion (Note a)**	**Comment**
Lightning	Y	C4	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. Both events are incorporated into the LSCS internal events model through the incorporation of generic and plant specific data. Based on this review, the Lightning impact hazard can be considered to be negligible.
Low Lake Level or River Stage	Y	C5	These effects would take place slowly allowing time for orderly plant reductions, including shutdowns. Based on this review, the Low Lake Level or River Stage impact hazard can be considered to be negligible.
Low Winter Temperature	Y	C4 C5	The principal effects of such events would be to cause a loss of off-site power. These effects would take place slowly allowing time for orderly plant reductions, including shutdowns (**C5**). At worst, the loss of off-site power events would be subsumed into the base PRA model results (**C4**). Based on this review, the Low Winter Temperature impact hazard can be considered to be negligible.
Meteorite or Satellite Impact	Y	PS4	The frequency of a meteor or satellite strike is judged to be so low as make the risk impact from such events insignificant. This hazard also was reviewed as part of the IPEEE submittal (Reference [45]) and screened based on low frequency of occurrence.

External Hazard	Screening Result		
	Screened? (Y/N)	**Screening Criterion (Note a)**	**Comment**
			Based on this review, the Meteorite or Satellite Impact impact hazard can be considered to be negligible.
Pipeline Accident	Y	C1	Per UFSAR Section 2.2.2.3 (Reference [41]), there are no tank farms or gas pipelines within 5 miles of the site. However, there are two natural gas pipelines between 5 and 7 miles from the site and two crude oil pipelines approximately 3 miles west of the plant. There is no significant hazard from toxic releases or explosions involving these pipelines that could interact with the plant. Based on this review, the Pipeline Accident impact hazard can be considered to be negligible.
Release of Chemicals in Onsite Storage	Y	C1	The impact of releases of hazardous materials stored on-site was evaluated in the IPEEE submittal and updated in LSCS station's UFSAR. UFSAR Section 2.2.3 (Reference [41]) discusses toxic gas. There is no onsite storage of chlorine; sodium hypochlorite/sodium bromide biocide system is used, thus eliminating an onsite chlorine hazard. Every 3 years a survey will be conducted to re-evaluate the use of chlorine, within 5 miles of the control room, to ensure that a chlorine hazard does not exist. Every 6 years a survey will be conducted to re-evaluate the use of toxic chemicals, within 5 miles of the control room, to ensure that a toxic chemical hazard does not exist. Based on this review, the Release of Chemicals in Onsite Storage impact

External Hazard	Screening Result		
	Screened? (Y/N)	Screening Criterion (Note a)	Comment
			hazard can be considered to be negligible.
River Diversion	Y	C1	Per UFSAR Section 2.4.9 (Reference [41]), the Illinois River flows in the same general location as its predecessor of nearly a million years ago. Presence of navigation locks and dams over the entire length of the river has further stabilized the river course. Based on the available evidence, no change in the regime of the river is expected. Based on this review, the River Diversion impact hazard can be considered to be negligible.
Sand or Dust Storm	Y	C1	The mid-western location of LSCS station precludes the possibility of a sandstorm. More common wind-borne dirt can occur but poses no significant risk to LSCS station given the robust structures and protective features of the plant. Based on this review, the Sand or Dust Storm impact hazard can be considered to be negligible.
Seiche	Y	C3	Flooding due to seiches is not relevant for LSCS station per Section 2.4.5 of the UFSAR [41]. Based on this review, the Seiche impact hazard can be considered to be negligible.
Seismic Activity	N/A	N/A	See Section 3.2.3 and Figure A4-1 in this Attachment.

External Hazard	Screening Result		
	Screened? (Y/N)	Screening Criterion (Note a)	Comment
Snow	Y	C5	This hazard is slow to develop and can be identified via monitoring and managed via normal plant processes. Potential flooding impacts covered under external flooding. Based on this review, the Snow impact hazard can be considered to be negligible.
Soil Shrink-Swell Consolidation	Y	C1	The potential for this hazard is low at the site, the plant design considers this hazard and the hazard is slow to develop and can be mitigated. Based on this review, the Soil Shrink-Swell Consolidation impact hazard can be considered to be negligible.
Storm Surge	Y	C3	The mid-western location of LSCS station precludes the possibility of a sea level driven storm surge. Based on this review, the Storm Surge impact hazard can be considered to be negligible.
Toxic Gas	Y	C3	UFSAR Section 2.2.3 (Reference [41]) discusses toxic gas. There is no onsite storage of chlorine; sodium hypochlorite/sodium bromide biocide system is used, thus eliminating an onsite chlorine hazard. In addition, there is no possibility of an accident that could lead to the formation of flammable clouds in the vicinity of LSCS because (1) there is no chemical plant in the vicinity; (2) no gas pipeline passes the station; and (3) no liquefied gases are transported in the vicinity.

External Hazard	**Screening Result**		
	Screened? (Y/N)	**Screening Criterion (Note a)**	**Comment**
			Per the IPEEE, the bounding analysis showed that these accidents do not significantly contribute to the plant risk. See also Transportation Accidents. Based on this review, the Toxic Gas impact hazard can be considered to be negligible.
Transportation Accident	Y	C1 C3 PS4	The impact of transportation accidents was evaluated in the IPEEE [45] and in UFSAR Section 2.2.3 [41]. In the IPEEE, an evaluation was conducted to demonstrate that the probability of a rail, land or waterway accident that resulted in release of toxic materials that could affect the site was less than 1E-6 /yr **(PS4)**. Per the UFSAR: <u>Flammable Vapor Clouds (delayed ignition):</u> There is no possibility of an accident that could lead to the formation of flammable clouds in the vicinity of LSCS because (1) there is no chemical plant in the vicinity; (2) no gas pipeline passes the station; and (3) no liquefied gases are transported in the vicinity **(C3)**. <u>Transportation of Toxic Chemicals:</u> The only transportation route carrying toxic chemicals which is within 5 miles of the station is the Illinois River. The toxic chemicals transported are chlorine and anhydrous ammonia. A toxic chemical analysis was performed (Reference [46]) which concluded that chlorine was an insignificant hazard to the station.

External Hazard	Screening Result		
	Screened? (Y/N)	Screening Criterion (Note a)	Comment
			For anhydrous ammonia, redundant detectors have been added on each outside air intake of the control room area filtration system. These detectors will sense ammonia concentrations at the outside air intakes from near zero ppm and higher. On detection of ammonia in the outside air, a control room annunciator alarms. Within 2 minutes of detection of high ammonia concentration in the air intake, the Operator will align the control room envelope HVAC systems in recirculation mode and will don a self-contained breathing apparatus. In accordance with the external hazard screening process per Figure 5-6 of NEI 00-04, the ammonia detectors and associated control room annunciators for ammonia were identified for categorization as High Safety Significant (HSS) SSCs should their associated systems be categorized. The ammonia detectors and associated control room annunciators are considered HSS for 50.69. **(C1)** <u>Explosions on the Highway:</u> For explosions on the highway, the worst event would be an explosion from a truck carrying 43,000 pounds of TNT on County Highway 6 at the nearest location to the plant (2000 feet away). If a 43,000-pound charge of TNT explodes at this distance, the structure will receive a peak reflected pressure of 1.5 psi. This magnitude is less than the tornado design pressure **(C1)**.

External Hazard	Screening Result		
	Screened? (Y/N)	Screening Criterion (Note a)	Comment
			Explosions on the Waterway: For explosions on the waterway, the volume of a maximum tank barge is about 1.8x10^5 ft^3. Assuming the air mix ratio is adequate for an empty gasoline barge and a detonation takes place, the energy released will be on the order of 107 kcal (Reference [47]), which is equivalent to an explosion of 10 tons of TNT. Since the Seismic Category I structures are located 4 miles away from the river, the peak reflected pressure on the structures will be less than 1 psi in case there is a detonation. Since the Seismic Category I structures have been designed for higher tornado wind pressures, the plant can withstand such a postulated explosion **(C1)**. Based on this review, the Transportation Accident impact hazard can be considered to be negligible.
Tsunami	Y	C3	The mid-western location of LSCS station precludes the possibility of a tsunami. Based on this review, the Tsunami impact hazard can be considered to be negligible.
Turbine-Generated Missiles	Y	C1	Per the IPEEE [45], the mean CDF for turbine-generated missiles was 1E-7/yr. Turbine generated missiles are discussed in UFSAR Section 3.5.1.3 (Reference [41]. With the replacement of the Low Pressure (LP) rotors, all the turbine rotors are of the monoblock design. The monoblock rotors have very low stress level. Missile generation due to turbine failure is

<table>
<tr><th rowspan="2">External Hazard</th><th colspan="3">Screening Result</th></tr>
<tr><th>Screened? (Y/N)</th><th>Screening Criterion (Note a)</th><th>Comment</th></tr>
<tr><td></td><td></td><td></td><td>generally postulated to be caused by turbine overspeed. General Electric has established that the speed capability of these rotors is considerably higher than the maximum attainable speed of these turbine generator units. Consequently, the probability of missiles being generated is statistically insignificant.

Based on this review, the Turbine-Generated Missiles impact hazard can be considered to be negligible.</td></tr>
<tr><td>Volcanic Activity</td><td>Y</td><td>C3</td><td>Not applicable to the site because of location (no active or dormant volcanoes located near plant site).

Based on this review, the Volcanic Activity impact hazard can be considered to be negligible.</td></tr>
<tr><td>Waves</td><td>Y</td><td>C3

C4</td><td>Waves associated with adjacent large bodies of water are not applicable to the site (C3). Waves associated with external flooding are covered under that hazard (C4).

Based on this review, the Waves impact hazard can be considered to be negligible.</td></tr>
<tr><td colspan="4">Note a – See Attachment 5 for descriptions of the screening criteria.</td></tr>
</table>

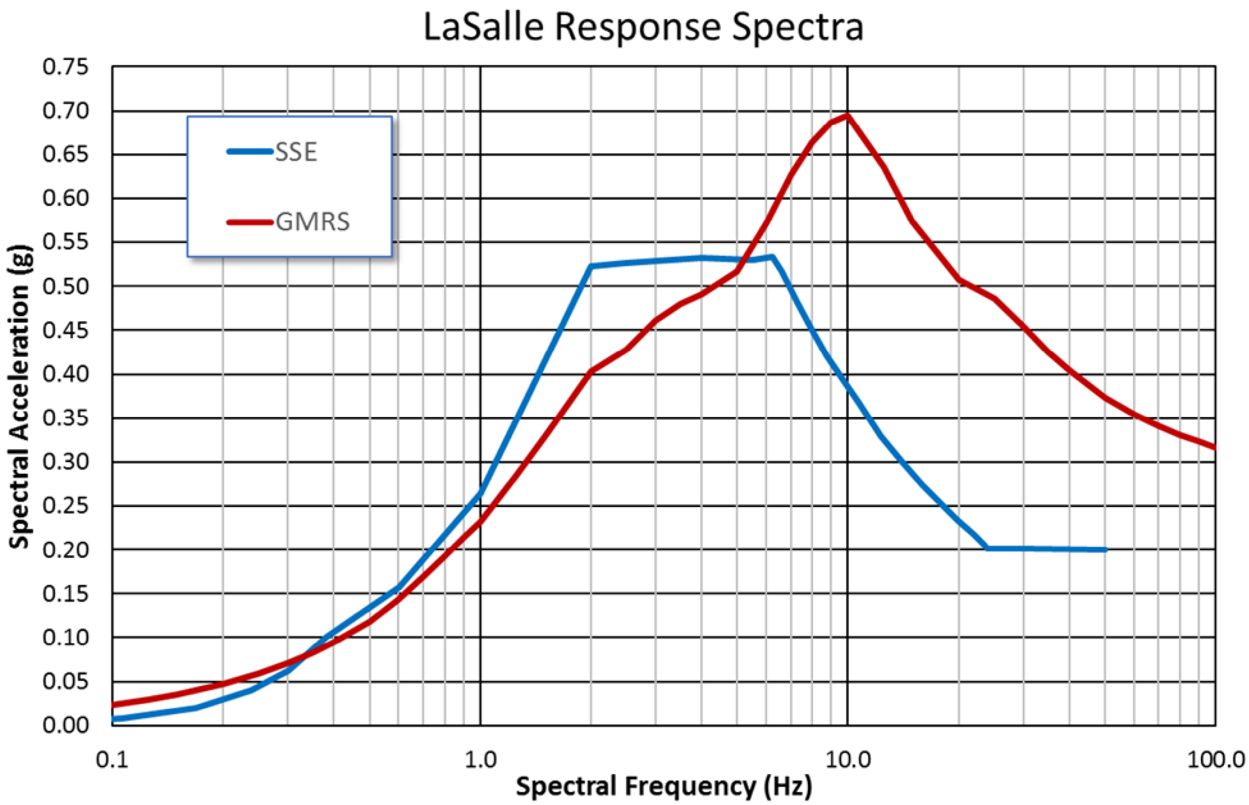

Figure A4-1: GMRS and SSE Response Spectra for LSCS
(From Reference [20] Sections 2.4 and 3.1

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	
	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	

Event Analysis	Criterion	Source	Comments
	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 Flooding PRA Model

In order to identify key sources of uncertainty for the 50.69 Program application, an evaluation of Internal Events baseline PRA model uncertainty was performed, based on the guidance in NUREG-1855 (Reference [48]) and Electric Power Research Institute (EPRI) report 1016737 (Reference [32]). As described in NUREG-1855, sources of uncertainty include "parametric" uncertainties, "modeling" uncertainties, and "completeness" (or scope and level of detail) uncertainties.

Parametric uncertainty was addressed as part of the LSCS County Generating Station (LSCS) baseline PRA model quantification (Reference [40]) and the Fire PRA uncertainty evaluation (Reference [49]).

Modeling uncertainties are considered in both the base PRA and in specific risk-informed applications. Assumptions are made during the PRA development as a way to address a particular modeling uncertainty because there is not a single definitive approach. Plant-specific assumptions made for each of the LSCS Internal Events PRA technical elements are noted in the individual notebooks. The Internal Events PRA model uncertainties evaluation is documented in Reference [40] and considers the modeling uncertainties for the base PRA by identifying assumptions, determining if those assumptions are related to a source of modeling uncertainty and characterizing that uncertainty, as necessary. EPRI compiled a listing of generic sources of modeling uncertainty to be considered for each PRA technical element (Reference [32]), and the evaluation performed for LSCS (Reference [40]) considered each of the generic sources of modeling uncertainty as well as the plant-specific sources.

Completeness uncertainty addresses scope and level of detail. Uncertainties associated with scope and level of detail are documented in the PRA but are only considered for their impact on a specific application (Reference [40]). 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.

Additionally, an evaluation of Level 2 internal events PRA model uncertainty was performed, based on the guidance in NUREG-1855 (Reference [48]) and Electric Power Research Institute (EPRI) report 1026511 (Reference [50]). The potential sources of model uncertainty in the LSCS PRA model were evaluated for the 32 Level 2 PRA topics outlined in EPRI 1026511.

A detailed review of the generic and plant-specific sources of internal events model uncertainties are discussed in LS-MISC-046 (Reference [51]) and are therefore not repeated in this attachment. The purpose of this attachment is to summarize the key sources of uncertainty that could potentially impact the 50.69 application.

Based on following the methodology in EPRI 1016737, as supplemented by EPRI 1026511, the impact of key sources of uncertainty in the internal events PRA model on the 50.69 application is summarized in Table 6-1. The key sources of uncertainty identified in Table 6-1 do not present a significant impact on the LSCS 50.69 application, and therefore, the internal events PRA model is capable of producing accurate 50.69 importance measure results.

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 uncertainty. The sensitivity studies are performed to ensure that assumptions and sources of uncertainty (e.g., human error, common cause failure, and maintenance probabilities) do not mask the SSC(s) importance. Regulatory Guide 1.174, Revision 3 (Reference [52]) cites NUREG-1855, Revision 1, as related guidance. In Section B of RG 1.174 (Reference [52]), the guidance acknowledges specific revisions of NUREG-1855 to include changes associated with expanding the discussion of uncertainties.

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

Table 6-1

IE / IF PRA Sources of Assumption/ Uncertainty	IE / IF PRA 50.69 Impact	IE / IF PRA Model Sensitivity and Disposition (50.69)
Core Melt Arrest Prior to Vessel Failure	Injection from these high capacity low pressure systems will preclude vessel failure if they are available following RPV depressurization given core damage occurs at high RPV pressure. ECCS survivability post containment venting is treated probabilistically. Although the treatment is realistic, there is the potential for a non-conservative bias given the unknown phenomenological events that could be associated with containment venting (e.g., hydrogen buildup in the Reactor Buildings, harsh events due to steam release, and other unknown consequences).	For this source of model uncertainty, sensitivity analyses were performed assuming that the ECCS is unavailable due to steam binding given failure to control containment venting. Operator actions related to containment venting remain the top risk-significant operator actions. The assumption is not realistic and use of this bounding failure probability would likely mask key risk insights. 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 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.

IE / IF PRA Sources of Assumption/ Uncertainty	IE / IF PRA 50.69 Impact	IE / IF PRA Model Sensitivity and Disposition (50.69)
		Therefore, the uncertainty associated with this model uncertainty is negligible within the 50.69 application.
Vapor Suppression Capabilities at Vessel Failure	Ex-vessel core melt progression overwhelms vapor suppression noted as extremely unlikely for low pressure RPV failures modes and very unlikely for high pressure failure modes based on reference to generic studies and identification of plant-specific features. However, more recent MAAP results indicate that containment pressurization following vessel failure for wet containment conditions might be higher than what had previously been calculated or what was originally considered.	Sensitivity analyses were performed using the recommended upper bound values from NUREG/CR-6595 (Reference [53]) for Mark II Containments as an alternate hypothesis (i.e., sensitivity analysis uses upper bound values of 0.2 for low pressure scenarios and 0.3 for high pressure scenarios). Operator actions related to containment venting remain the top risk-significant operator actions. The bounding sensitivity analysis utilizes the upper bound values, which is not a realistic assumption. 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 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. Therefore, the uncertainty associated with this model uncertainty is negligible within 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)
Digital Feedwater Controls	There are model uncertainties associated with modeling digital systems, such as those related to determining the failure modes of these systems and components. The reliability values from the similar vendor study demonstrating that the system performance would result in less than 0.1 transients per year are used for the key components of the system. The reliability analysis for causing plant trips performed by similar FW vendor studies is assumed to be equally applicable to the reliability of the system post plant trips that are caused by other means that do not directly affect the feedwater availability.	The sensitivity analyses consisted of increasing the failure probability associated with digital feedwater controls by a factor of 50 (i.e., from 0.01 to 0.5). The results demonstrate that the digital feedwater controls failure probability does not significantly impact the overall average-maintenance PRA results. Due to the small impact demonstrated by the sensitivity cases, the uncertainty associated with this model uncertainty is negligible.
Water Hammer Pipe Rupture	Water hammer is a potential failure mode of important systems and can also cause a flood related event. ECCS system draindown scenarios are included in the LSCS PRA model. Subsequent starting or restarting of these systems causes a water hammer and system leak or rupture.	The sensitivity analyses consisted of increasing the ECCS pipe rupture failure probabilities due to water hammers by a factor of 100 (i.e., from 1E-3 to 1E-1). The sensitivity analysis is only increasing the likelihood of pipe rupture due to water hammer events and additional equipment / accident mitigation strategies remain unaffected (i.e., the sensitivity analysis does not postulate additional equipment being out-of-service / unavailable). Due to the small impact demonstrated by the sensitivity cases, the uncertainty associated with this model uncertainty is negligible. This sensitivity

IE / IF PRA Sources of Assumption/ Uncertainty	IE / IF PRA 50.69 Impact	IE / IF PRA Model Sensitivity and Disposition (50.69)
		analysis assumes the upper bound pipe rupture failure probability for ECCS, which would not be realistic and use of this bounding rupture failure probability would likely mask key risk insights.

Fire PRA Model

The purpose of the following discussion is to address the epistemic uncertainty in the LSCS FPRA. The LSCS FPRA model includes various sources of uncertainty that exist because there is both inherent randomness in elements that comprise the FPRA and because the state of knowledge in these elements continues to evolve. The development of the LSCS FPRA was guided by NUREG/CR-6850 (Reference [54]. The LSCS FPRA model used consensus models described in NUREG/CR-6850.

LSCS used guidance provided in NUREG/CR-6850 and NUREG-1855 (Reference [48]) to address uncertainties associated with FPRA for the 50.69 program application. As stated in Section 1.3 of NUREG-1855:

> *"Although the guidance in this report does not currently address all sources of uncertainty, the guidance provided on the uncertainty identification and characterization process and on the process of factoring the results into the decision making is generic and independent of the specific source of uncertainty. Consequently, the guidance is applicable for sources of uncertainty in PRAs that address at-power and low power and shutdown operating conditions, and both internal and external hazards."*

NUREG-1855 also describes an approach for addressing sources of model uncertainty and related assumptions. It defines:

> *"A source of model uncertainty exists when (1) a credible assumption (decision or judgment) is made regarding the choice of the data, approach, or model used to address an issue because there is no consensus and (2) the choice of alternative data, approaches or models is known to have an impact on the PRA model and results. An impact on the PRA model could include the introduction of a new basic event, changes to basic event probabilities, change in success criteria, or introduction of a new initiating event. A credible assumption is one submitted by relevant experts and which has a sound technical basis. Relevant experts include those individuals with explicit knowledge and experience for the given issue. An example of an assumption related to a source of model uncertainty is battery depletion time. In calculating the depletion time, the analyst may not have any data on the time required to shed loads and thus may assume (based on analyses) that the operator is able to shed certain electrical loads in a specified time."*

NUREG-1855 defines consensus model as:

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

The plant-specific assumptions in the LSCS FPRA (Reference [49]) and the 71 generic sources of uncertainty identified in EPRI 1026511 (Reference [50]) were evaluated for their potential impact

on the 50.69 application. This guideline organizes the uncertainties in Topic Areas similar to those outlined in NUREG/CR-6850 and was used to evaluate the baseline FPRA epistemic uncertainty and evaluate the impact of this uncertainty on 50.69 SSC component importance measures.

A detailed review of the generic and plant-specific sources of internal fire model uncertainties are discussed in LS-MISC-046 (Reference [51]) and are therefore not repeated in this attachment. The purpose of this attachment is to summarize the key sources of uncertainty that could potentially impact the 50.69 application.

Table 6-2 summarizes the review for key sources of uncertainty in the internal fire PRA model for the 50.69 application (organized by NUREG/CR-6850 tasks).

As noted above, the LSCS FPRA was developed using consensus methods outlined in NUREG/CR-6850 and interpretations of technical approaches as required by NRC. Fire PRA methods were based on NUREG/CR-6850, other more recent NUREGs, (e.g., NUREG-7150 (Reference [55]), and published "frequently asked questions" (FAQs) for the Fire PRA.

The key sources of uncertainty identified in Table 6-2 do not present a significant impact on the LSCS 50.69 application, and therefore, the fire PRA model is capable of producing accurate 50.69 importance measure results.

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 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. Regulatory Guide 1.174, Revision 3 (Reference [52]) cites NUREG-1855, Revision 1, as related guidance. In Section B of RG 1.174 (Reference [52]), the guidance acknowledges specific revisions of NUREG-1855 to include changes associated with expanding the discussion of uncertainties.

Table 6-2 below describes the fire PRA sources of model uncertainty and their impact.

Table 6-2

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.	Based on a review of the assumptions and potential sources of sources of uncertainly associated with this element, it is concluded that the methodology for the Analysis Boundary and Partitioning task does not introduce any epistemic uncertainties that would affect the 50.69 program. Therefore, this does not represent a key source of uncertainty for the LSCS 50.69 application.

Fire PRA Description	Fire PRA Sources of Uncertainty	Fire PRA Disposition
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.	The uncertainty associated with this task is related to the identification of all components that should be credited/linked in the FPRA. This source of uncertainty is reduced as a result of multiple overlapping tasks including the MSO expert panel, reviews of FPIE screened initiating events, screened containment penetrations, and screened ISLOCA scenarios. Additional internal reviews of analysis results further reduce the uncertainty associated with this task. Based on a review of the assumptions and potential sources of uncertainty related to this element and the discussion above, it is concluded that the methodology for the Component Selection task does not introduce any epistemic uncertainties that would affect the 50.69 program. Therefore, this does not represent a key source of uncertainty for the LSCS 50.69 application.
Cable Selection	The selection of cables to be considered in the analysis is identified using industry guidance documents. The overall process is essentially the same as that used to perform the analyses to demonstrate compliance with 10 CFR 50.48.	Additionally, as part of the Fire PRA, some components were conservatively assumed to be failed based on lack of cable data. Components in this category are referred to as Unknown Location (UNL) components because specific cables were not identified for the components. Based on recent Fire PRA updates, the UNL components are mostly limited to Balance of Plant (BOP) systems. A sensitivity analysis was performed to measure the risk associated with the assumption that these components fail in select fire scenarios. The sensitivity

Fire PRA Description	Fire PRA Sources of Uncertainty	Fire PRA Disposition
		removed all UNL components from every fire scenario, as described in the Uncertainty & Sensitivity Analysis Notebook. Based on the results, the inclusion of the UNL components introduces moderate risk to both Fire CDF and LERF. Although the sensitivity shows a moderate impact on Fire CDF and Fire LERF, complete removal of UNLs would not be considered realistic since those cables could be identified with detailed circuit analysis and those failures would exist in specific areas of the plant. Also, the dominant fire scenarios are undeveloped full room burnouts that when refined with detailed fire modeling and fire scenario development would reduce the overall impact of the bounding sensitivity. Given that an informed approach was used to developing the assumed routing, the methodology employed by the Fire PRA is appropriate. Based on a review of the assumptions and potential sources of uncertainty related to this element it is concluded that the methodology for the Cable Selection task does not introduce any epistemic uncertainties that would affect the 50.69 program. Therefore, this does not represent a key source of uncertainty for the LSCS 50.69 application.
Qualitative Screening	Qualitative screening was 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	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.

Fire PRA Description	Fire PRA Sources of Uncertainty	Fire PRA Disposition
	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.	Based on a review of the assumptions and potential sources of uncertainty related to this element and the discussion above, it is concluded that the methodology for the Qualitative Screening task does not introduce any epistemic uncertainties that would affect the 50.69 program. Therefore, this does not represent a key source of uncertainty for the LSCS 50.69 application.
Fire-Induced Risk Model	The internal events PRA model was updated to add fire specific initiating event structure as well as additional system logic. The methodology used is consistent with that used for the internal events PRA model development and was subjected to industry Peer Review. The developed model is applied in such a fashion that all postulated fires are assumed to generate a plant trip. This represents a source of uncertainty, as it is not necessarily clear that fires would result in a trip. In the event the fire results in damage to cables and/or equipment identified in Task 2, the PRA model includes structure to translate them into the appropriate induced initiator.	The identified source of uncertainty could result in the over-estimation of fire risk. In general, the Fire PRA development process would have reviewed significant fire initiating events and performed supplemental assessments to address this possible source of uncertainty. Based on a review of the assumptions and potential sources of uncertainty related to this element and the discussion above, it is concluded that the methodology for the Fire-Induced Risk Model task does not introduce any epistemic uncertainties that would affect the 50.69 program. Therefore, this does not represent a key source of uncertainty for the LSCS 50.69 application.
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. The primary source of uncertainty	The LSCS Fire PRA utilized the bin frequencies from NUREG/CR-2169 (Reference [56]), which represents the most current approved source for bin frequencies. As such, some of the inherent conservatism associated with bin frequencies from NUREG/CR-6850 was removed. A parametric uncertainty analysis using the Monte Carlo

Fire PRA Description	Fire PRA Sources of Uncertainty	Fire PRA Disposition
	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. LSCS uses the ignition frequencies in NUREG-2169 (Reference [56]) along with the revised heat release rates from NUREG - 2178 (Reference [57]).	method is provided in the FPRA documentation. Consensus approaches are employed in the model. Based on a review of the assumptions and potential sources of uncertainty related to this element it is concluded that the methodology for the Fire Ignition Frequency task does not introduce any epistemic uncertainties that would affect the 50.69 program. Therefore, this does not represent a key source of uncertainty for the LSCS 50.69 application.
Quantitative Screening	Other than screening out potentially risk significant scenarios (ignition sources), this task is not a source of uncertainty.	Quantitative screening criteria was defined for the LSCS Fire PRA as the CDF / LERF contribution of zero, such that all quantified fire scenarios are retained. All of the results were retained in the cumulative CDF / LERF, therefore, no uncertainty was introduced as a result of this task. Based on the discussion above, it is concluded that the methodology for the Quantitative Screening task does not introduce any epistemic uncertainties that would affect the 50.69 program. Therefore, this does not represent a key source of uncertainty for the LSCS 50.69 application.
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 discussion.

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 Fire PRA were performed on a case-by-case basis where the scenario risk quantification was large enough to warrant further detailed analysis. Hot short probabilities and hot short duration probabilities as defined in NUREG-7150, Volume 2, based on actual fire test data, were used in the LSCS Fire PRA. The uncertainty (conservatism) which may remain in the Fire PRA is associated with scenarios that do not contribute significantly to the overall fire risk. Based on a review of the assumptions and potential sources of uncertainty related to this element and the discussion above, it is concluded that the methodology for the Detailed Circuit Failure Analysis task does not introduce any epistemic uncertainties that would affect the 50.69 program. Therefore, this does not represent a key source of uncertainty for the LSCS 50.69 application.
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 and a hot short duration probability are assigned using industry guidance published in NUREG-7150, Volume 2 (Reference [58]). The uncertainty values specified in NUREG-7150, Volume 2 are based on fire test data.	The use of hot short failure probability and duration probability is based on fire test data and associated consensus methodology published in NUREG-7150, Volume 2 (Reference [58]). Based on a review of the assumptions and potential sources of uncertainty related to this element and the discussion above, it is concluded that the methodology for the Circuit Failure

Fire PRA Description	Fire PRA Sources of Uncertainty	Fire PRA Disposition
		Mode Likelihood Analysis task does not introduce any epistemic uncertainties that would affect the 50.69 program. Therefore, this does not represent a key source of uncertainty for the LSCS 50.69 application.
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 response of plant staff (detection, fire control, fire suppression). The fire modeling methodology itself is largely empirical in some 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	Consensus modeling approach is used for Detailed Fire Modeling and it is concluded that the methodology for the Detailed Fire Modeling task does not introduce any epistemic uncertainties that would require sensitivity treatment. Therefore, this does not represent a key source of uncertainty for the LSCS 50.69 application.

Fire PRA Description	Fire PRA Sources of Uncertainty	Fire PRA Disposition
	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	The Human Error Probabilities (HEPs) used in the FPRA were adjusted to consider the additional challenges that may be present given a fire. The HEPs included the consideration of degradation or loss of necessary cues due to fire. Given the methodology used, the impact of any remaining uncertainties is expected to be small.	The HEPs include the consideration of degradation or loss of necessary cues due to fire. The fire risk importance measures indicate that the results are somewhat sensitive to HRA model and parameter values. The LSCS Fire PRA model HRA 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 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. It is concluded that the methodology for the Post-Fire Human Reliability Analysis task does not introduce any epistemic uncertainties that would require sensitivity treatment. Therefore, this does not represent a key source of uncertainty for the LSCS 50.69 application.

Fire PRA Description	Fire PRA Sources of Uncertainty	Fire PRA Disposition
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 uncertainty as it is not expected to provide changes to the quantified Fire PRA model. Based on the discussion above, it is concluded that the methodology for the Seismic-Fire Interactions Assessment task does not introduce any epistemic uncertainties that affect the 50.69 program. Therefore, this does not represent a key source of uncertainty for the LSCS 50.69 application.
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.	The selected truncation was confirmed to be consistent with the requirements of the PRA Standard (Reference [59]). Based on a review of the assumptions and potential sources of uncertainty related to this element and the discussion above, it is concluded that the methodology for the Fire Risk Quantification task does not introduce any epistemic uncertainties that would affect the 50.69 program. Therefore, this does not represent a key source of uncertainty for the LSCS 50.69 application.
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. Additionally, for the 50.69 program, the guidance in NEI 00-04 (Reference [1]) specifies that certain sensitivity studies be

Fire PRA Description	Fire PRA Sources of Uncertainty	Fire PRA Disposition
		conducted for each PRA model to address key sources of 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. Based on the discussion above, it is concluded that the methodology for the Uncertainty and Sensitivity Analyses task does not introduce any epistemic uncertainties that would affect the 50.69 program. Therefore, this does not represent a key source of uncertainty for the LSCS 50.69 application.
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. Based on the discussion above, it is concluded that the methodology for the Fire PRA documentation task does not introduce any epistemic uncertainties that would affect the 50.69 program. Therefore, this does not represent a key source of uncertainty for the LSCS 50.69 application.

Attachment 7: Comparison of RG 1.200 Revision 1 and Revision 2 SRs Applicable to CC-I/II, CC-II/III, and CC-I/II/III

Finding Number	Supporting Requirement(s)	Capability Category (CC)	Description	Disposition for 50.69
URE LS2020-0001 (Update Requiring Evaluation Tracking ID)	IFSO-A3 IFSN-A7 IFQU-A3	Not Met CCII	As part of the Self-Assessment performed during the 2014 FPIE PRA update, the following gaps to RG 1.200 (Rev. 2) and the ASME/ANS PRA Standard were identified: 1. IFSO-A3 Further documentation clarification is required for those flood locations that are screened out based on the quantitative screening criteria described in the PRA Standard. 2. IFSN-A7 Further documentation clarification is required for justification of crediting EQ limits for ensuring operability of instrumentation given spray-induced impacts. 3. IFQU-A3 Further documentation clarification is required for those flood locations that are screened out based on the quantitative screening criteria described in the PRA Standard.	Open These open issues have no impact on the 50.69 implementation as they are primarily documentation issues. However, these gaps will be resolved during a PRA update.