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10 CFR 50.69
10 CFR 50.90

RS-23-060

June 8, 2023

U.S. Nuclear Regulatory Commission
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Quad Cities Nuclear Power Station, Units 1 and 2
Renewed Facility Operating License Nos. DPR-29 and DPR-30
NRC Docket Nos. 50-254 and 50-265

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 10 CFR 50.90, "Application for amendment of license, construction permit, or early site permit," Constellation Energy Generation, LLC (CEG) requests an amendment to Renewed Facility Operating License Nos. DPR-29 and DPR-30 for Quad Cities Nuclear Power Station (QCNPS) Units 1 and 2.

The proposed amendment modifies the QCNPS licensing basis, by the addition of a License Condition, to implement 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 either not be changed or will be enhanced. This allows improved focus on equipment that has high safety significance resulting in improved plant safety.

The enclosure to this letter provides the basis for the proposed change to the QCNPS Facility Operating License. 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 QCNPS submittal RS-23-059 dated June 8, 2023, "Application to Revise Technical Specifications to Adopt Risk Informed Completion Times TSTF-505, Revision 2, 'Provide Risk-Informed Extended Completion Times – RITSTF Initiative 4b'." CEG 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 enhance the efficiency of CEG and NRC resources used for the review of the applications. These requests should not be considered linked licensing actions, as the details of the PRA models in each LAR are complete which will allow the NRC to independently review and approve each LAR on its own merits without regard to the results from the review of the other.

CEG requests approval of the proposed change by June 7, 2024. The amendment shall be implemented within 60 days of approval.

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), CEG 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 in this letter. Should you have any questions concerning this letter, please contact Ms. Rebecca L. Steinman at (630) 657-2831.

I declare under penalty of perjury that the foregoing is true and correct. Executed on the 8th day of June 2023.

Respectfully,

Patrick R. Simpson
Sr. Manager Licensing
Constellation Energy Generation, LLC

Enclosure: Application to Adopt 10 CFR 50.69

Attachments to the Enclosure:

1. List of Categorization Prerequisites
2. Description of PRA Models Used in Categorization
3. Disposition and Resolution of Open Peer Review Findings and Self-Assessment Open Items
4. External Hazards Screening
5. Progressive Screening Approach for Addressing External Hazards
6. Disposition of Key Assumptions/Sources of Uncertainty

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

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LIST OF ATTACHMENTS

1 SUMMARY DESCRIPTION

The proposed amendment modifies the Quad Cities Nuclear Power Station (QCNPS) licensing basis, by the addition of a License Condition, 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 either not be changed or will be enhanced. This allows improved focus on equipment that has high safety significance resulting in improved plant safety.

2 DETAILED DESCRIPTION

2.1 Current Regulatory Requirements

The U.S. 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, 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 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 using 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 (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 either not be changed or will be enhanced. This allows improved focus on equipment that has high 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 assure functionality and reliability are maintained and is a function of the SSC categorization results and associated bases. Finally, periodic assessment activities are conducted to adjust the categorization and/or treatment processes as needed so that SSCs continue to meet all applicable 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 HSS, existing treatment requirements are maintained or enhanced. Conversely, for SSCs that do not significantly contribute to plant safety on an individual basis, the rule allows an alternative risk-informed approach to treatment that provides reasonable confidence (which is a reduced level compared to the reasonable assurance criteria used for many special treatments) that these SSCs will satisfy functional requirements.

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

2.3 <u>Description of the Proposed Change</u>

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

> Constellation Energy Generation, LLC is approved to implement 10 CFR 50.69 using the processes for categorization of Risk-Informed Safety Class (RISC)-1, RISC-2, RISC-3, and RISC-4 structures, systems, and components (SSCs) using: Probabilistic Risk Assessment (PRA) models to evaluate risk associated with internal events, including internal flooding, and internal fire; the shutdown safety assessment process to assess shutdown risk; the Arkansas Nuclear One, Unit 2 (ANO-2) passive categorization method to assess passive component risk for Class 2 and Class 3 and non-Class SSCs and their

associated supports; the results of the non-PRA evaluations that are based on the IPEEE Screening Assessment for External Hazards updated using the external hazard screening significance process identified in ASME/ANS PRA Standard RA-Sa-2009 for other external hazards except seismic; and the alternative seismic approach as described in Constellation's submittal letter dated June 8, 2023, 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 individually addressed in Sections 3.1 through 3.4 below.

The PRA models described within this license amendment request (LAR) are the same as those described within the CEG submittal RS-23-059, "License Amendment Request to Revise Technical Specifications to Adopt Risk Informed Completion Times TSTF-505, Revision 2, 'Provide Risk-Informed Extended Completion Times – RITSTF Initiative 4b,'" dated June 8, 2023 (Reference [2]).

CEG requests that the NRC conduct their review of the PRA technical adequacy details for this application in coordination with the review of the TSTF-505 (RICT Program) application currently in-process. Concurrent review will enhance efficiencies for the use of CEG and NRC resources necessary to complete the review of the separate applications. These requests should not be considered linked requested licensing actions (RLAs), since each LAR independently captures the complete details of the PRA models which would allow the NRC to independently review and approve each LAR on its 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

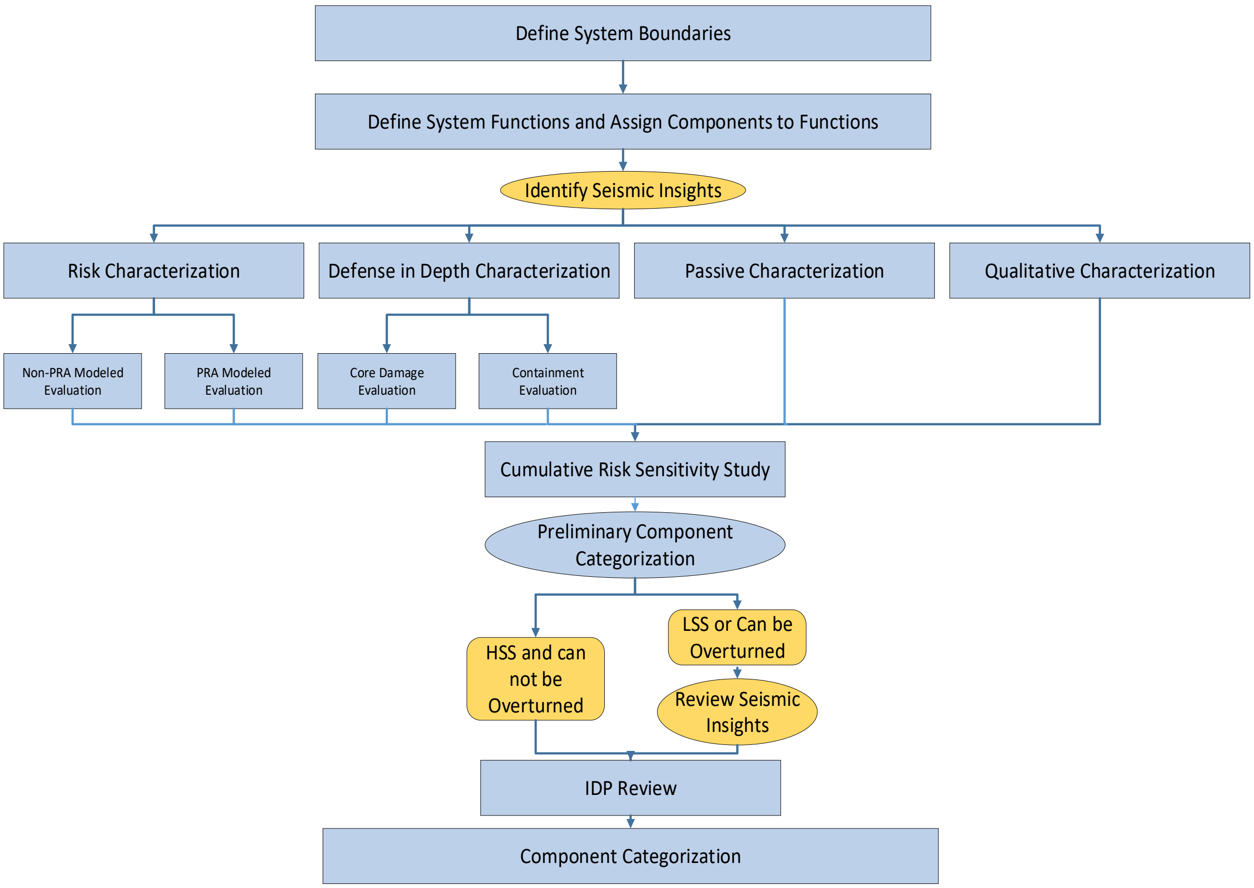
CEG intends to implement a risk categorization process at QCNPS in accordance with NEI 00-04, Revision 0, "10 CFR 50.69 SSC Categorization Guideline," as endorsed by Regulatory Guide (RG) 1.201, "Guidelines for Categorizing Structures, Systems, and Components in Nuclear Power Plants According to their Safety Significance" (Reference [3]). 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, as endorsed by RG 1.201, with the exception of the seismic hazard impact evaluation, which will use the Electric Power Research Institute (EPRI) 3002017583 (Reference [4]) approach for seismic Tier 1 sites, which includes QCNPS, to assess seismic hazard risk for 50.69. Inclusion of additional process steps discussed below to address seismic considerations will ensure that the reasonable confidence 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) are completed is flexible 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 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. Defense-in-depth assessment
5. Passive categorization methodology

Figure 3-1 illustrates the major steps of the categorization process described in NEI 00-04. As explained further in Section 3.2.3, two categorization steps (represented by four blocks on the figure) have been included to highlight review of seismic insights as they pertain to this application.

Figure 3-1: Categorization Process Overview



Categorization of SSCs at QCNPS 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 that the term "preliminary HSS or LSS" used in this application 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 the NEI 00-04 IDP limitations. The steps of the process are performed at either the function level, component level, or both. This is also summarized in 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-PRA Modeled)	Fire and Other External Hazards	Component	Not Allowed	No
	Seismic	Function/Component	Allowed [2]	No
	Shutdown – Section 5.5	Function/Component	Not Allowed	No
Defense-in-Depth	Core Damage – Section 6.1	Function/Component	Not Allowed	Yes
	Containment – Section 6.2	Component	Not Allowed	Yes
Qualitative Criteria	Considerations – Section 9.2	Function	Allowable[1]	N/A
Passive	Passive – Section 4	Segment/Component	Not Allowed	No

Notes:

[1] The assessments of the qualitative considerations are agreed upon by the IDP in accordance with Section 9.2. In some cases, a 10 CFR 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 10 CFR 50.69 categorization team for at least the system functions that are not found to be HSS by 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 10 CFR 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 10 CFR 50.69 team (i.e., all considerations for all functions are reviewed). The IDP may confirm the preliminary function risk and associated justification or may direct that it be changed based upon their expert knowledge. 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. Additionally, Section 4.0 discusses additional functions that may be identified (e.g., fill and drain) to group and reconsider potentially LSS components that may have been initially associated with a HSS function but 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 or may remain LSS. Given that QCNPS is a seismic Tier 1 (low seismic hazard) plant as defined in Reference [4], 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 clarifications are 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, (1) the purpose of the categorization; (2) present treatment requirements for SSCs including requirements for design basis events; (3) PRA fundamentals; (4) 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 (5) the defense-in-depth philosophy and requirements to maintain this philosophy.

- The decision criteria for the IDP to categorize SSCs as safety significant or LSS in accordance with § 50.69(f)(1) will be documented in CEG 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 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 [5]) 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.

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

- QCNPS proposes to apply an alternative seismic approach to those listed in NEI 00-04 Sections 1.5 and 5.3. This approach is specified in EPRI 3002017583 (Reference [4]) for Tier 1 plants and is discussed in Section 3.2.3 of this enclosure.

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 June 8, 2023 – See Reference [2] Enclosure 5 and Attachment 2.

- Fire Risks: Fire PRA model, as submitted to the NRC for TSTF-505 dated June 8, 2023 – See Reference [2] Enclosure 5 and Attachment 2.

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

- Other External Risks (e.g., tornados, external floods): Using the Individual Plant Examination of External Events (IPEEE) screening process as approved by NRC SE dated April 26, 2001, (TAC Nos. M83665 and M83666) (Reference [6]). 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 [7]), 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 [8] consistent with the related Safety Evaluation (SE) issued by NRC.

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

The use of this method for a 10 CFR 50.69 application was previously approved in the final SE for Vogtle dated December 17, 2014 (Reference [5]). The RI-RRA method as approved for use at Vogtle for 10 CFR 50.69 does not have any plant specific aspects and is therefore 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 since 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 RG 1.147, Revision 15 (Reference [9]). 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 HSS for passive categorization which will result in HSS for its risk-informed safety classification that cannot be changed by the IDP. Therefore, this methodology and scope for passive categorization is acceptable and appropriate for use at QCNPS 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 described within this document are the same as those described within QCNPS submittal RS-23-059 dated June 8, 2023, "Application to Revise Technical Specifications to Adopt Risk Informed Completion Times TSTF-505, Revision 2, 'Provide Risk-Informed Extended Completion Times – RITSTF Initiative 4b'" (Reference [2]).

3.2.1 Internal Events and Internal Flooding

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

3.2.2 Fire Hazards

The QCNPS 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 (Reference [10]) and only utilizes methods previously accepted by the NRC. The CEG risk management process ensures that the PRA model used in this application reflects the as-built and as-operated plant for QCNPS. Attachment 2 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 [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 QCNPS seismic hazard assessment, CEG 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 EPRI 3002017583, "Alternative Approaches for Addressing Seismic Risk in 10 CFR 50.69 Risk-Informed Categorization" (Reference [4]), and includes additional qualitative considerations that are discussed in this section[1].

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

> *Tier 1: Plants where the GMRS [Ground Motion Response Spectrum] peak acceleration is at or below approximately 0.2g or where the GMRS is below or approximately equal to the SSE [Safe Shutdown Earthquake] between 1.0 Hz and 10 Hz. Examples are shown in Figures 2-1 and 2-2. At these sites, the GMRS is either very low or within the range of the SSE such that unique seismic categorization insights are not expected.*
>
> Note: EPRI 3002017583 applies to the Tier 1 sites in its entirety except for Sections 2.3 (Tier 2 sites), 2.4 (Tier 3 sites), Appendix A (seismic correlation), and Appendix B (criteria for capacity-based screening).

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

The trial studies in EPRI 3002017583 show that seismic categorization insights are overlaid by other risk insights even at plants where the GMRS is far beyond the seismic design basis. Therefore, the basis for the Tier 1 classification and resulting criteria is not that the design basis insights are adequate. Instead, it is that consideration of the full range of the seismic hazard

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[1] EPRI 3002017583 is an update to EPRI 3002012988, "Alternative Approaches for Addressing Seismic Risk in 10 CFR 50.69 Risk-Informed Categorization," July 2018 (Reference [73]) which was referenced in the NRC issued amendment and SE for Calvert Cliffs Nuclear Power Plant, Units 1 and 2, to implement 10 CFR 50.69 as noted below:

(1) Calvert Cliffs Nuclear Power Plant, Units 1 and 2, "Issuance of Amendment Nos. 332 and 310 Re: Risk-Informed Categorization and Treatment of Systems, Structures, and Components (EPID L-2018-LLA-0482)," February 28, 2020 (ADAMS Accession No. ML19330D909) (Reference [82]).

(2) This license amendment request incorporates by Reference the Clinton Power Station, Unit 1 response to request for additional information letter of November 24, 2020 (ML20329A433) (Reference [83]), in particular, the response to the question regarding the differences between the EPRI 3002012988 (initial) and EPRI 3002017583 (current) as well as CEG's proposed approach for the 50.69 Seismic Alternative Tier 1.

produces limited unique insights to the categorization process. That is the basis for the following statements in Table 4-1 of the EPRI report.

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

The proposed categorization approach for QCNPS is a risk-informed graded approach that is demonstrated to produce categorization insights equivalent to a seismic PRA. For Tier 1 plants, this approach relies on the insights gained from the seismic PRAs examined in Reference [4] along with confirmation that the site GMRS is low. Reference [4] demonstrates that seismic risk is adequately addressed for Tier 1 sites by the results of additional qualitative assessments discussed in this section and existing elements of the 10 CFR 50.69 categorization process specified in NEI 00-04.

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

EPRI 3002017583 recommends a risk-informed graded approach for addressing the seismic hazard in the 10 CFR 50.69 categorization process. There are a number of seismic fragility fundamental concepts that support a graded approach and there are important characteristics about the comparison of the seismic design basis (represented by the SSE) to the site-specific seismic hazard (represented by the GMRS) that support the selected thresholds between the three evaluation tiers in the EPRI report. The coupling of these concepts with the categorization process in NEI 00-04 are the key elements of the approach defined in EPRI 3002017583 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 [11]) 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. At sites with lower seismic demands such as QCNPS, there is no need to perform more detailed evaluations to demonstrate the inherent seismic capacities documented in industry sources such as Reference [11]. Low seismic demand sites have lower likelihood of seismically-induced failures and lesser challenges to plant systems, providing the technical basis for allowing use of a graded approach for addressing seismic hazard at QCNPS.

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

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

a. SSCs for which the inherent seismic capacities are applicable, or which are designed to the plant SSE will have low probabilities of failure at sites where the peak spectral acceleration of the GMRS < 0.2g or where the GMRS < SSE between 1 and 10 Hz.
b. The low probabilities of failure of individual components would also apply to components considered to have correlated seismic failures.
c. These low probabilities of failure lead to low seismic CDF and LERF estimates from an absolute risk perspective.
d. The low seismic CDF and LERF estimates lead to reasonable confidence that seismic risk contributions would allow reducing a HSS to LSS due to the 10 CFR 50.69 Integral Assessment provided the equipment is HSS only due to seismic considerations.

Test cases described in Section 3 of Reference [4] showed that it would be unusual, even for moderate hazard plants, to exhibit any unique seismic insights, including one's due to correlated failures. CEG is using other licensees' plant-specific test case information (specifically, Reference [4] Case Study A, C, and D information) and incorporating that information by reference into this application. NRC review and acceptance of the seismic PRA corresponding to these test cases is described in References [12] through [13]. While it is prudent to perform additional evaluations to identify conditions where correlated failures may occur for Tier 2 sites, for Tier 1 sites such as QCNPS, correlation studies would not lead to new seismic insights or affect the baseline seismic CDF in any significant way.

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

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

In response to the NRC 50.54(f) letter associated with post-Fukushima recommendations (Reference [14]), QCNPS submitted a seismic hazard screening report (Reference [15]) to the NRC. The Golden Gate curve dips slightly below the GMRS between 1-1.5 Hz and the Housner curve dips slightly below the new GMRS curve between 7-10 Hz; however, both curves supplement each other and bound the new GMRS in the 1 Hz to 10 Hz range. As a result, QCNPS meets the Tier 1 criterion in Reference [4].

The QCNPS SSE and GMRS curves from the seismic hazard and screening response in Reference [15] are shown in Figure A4-1. The NRC's assessment of the QCNPS seismic hazard and screening response is documented in Reference [16]. In Section 3.4 of Reference [16] the NRC concluded that the methodology used by QCNPS in determining the GMRS was acceptable and that the GMRS determined by QCNPS adequately characterizes the reevaluated hazard for the QCNPS site. In addition, Reference [16] in Section 4.0 confirms the licensee's conclusion that the licensee's GMRS for the QCNPS site is bounded by the combination of design spectrum defining the SSE over the 1 to 10 Hz frequency range.

Section 1.1.3 of Reference [4] cites various post-Fukushima seismic reviews performed for the U.S. fleet of nuclear power plants. The docketed seismic reviews prepared by the licensee and the corresponding NRC assessments applicable to QCNPS are listed here.

1. NTTF Recommendation 2.1 seismic hazard screening (References [15], [17], and [16]).
2. NTTF Recommendation 2.3 seismic walkdowns (References [18] and [19]).
3. NTTF Recommendation 4.2 seismic mitigation strategy assessment (S-MSA) (References [20] and [21]).

The following additional post-Fukushima seismic reviews were performed for QCNPS.

4. NTTF Recommendation 2.1 seismic high frequency evaluation (References [22] and [23]).

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

As an enhancement to the EPRI study results as they pertain to QCNPS, the proposed QCNPS categorization approach for seismic hazards will include qualitative consideration of the mitigation capabilities of SSCs during seismically-induced events and seismic failure modes, based on insights obtained from prior seismic evaluations performed for QCNPS. For example, the System Categorization Document (SCD) that is presented to the IDP will include a section that summarizes the identified plant seismic insights pertinent to the system being categorized and will also state the basis for applicability of the EPRI 3002017583 study as well as the bases for QCNPS being a Tier 1 plant. The discussion of the Tier 1 bases will include such factors as:

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

At several steps of the categorization process (e.g., as noted in Figure 3-1 and Table 3-1) the categorization team will consider the available seismic insights relative to the system being categorized and document their conclusions in the SCD. Integrated importance measures over all modeled hazards (i.e., internal events, including internal flooding, and internal fire for QCNPS) 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.

HSS SSCs uniquely identified by the QCNPS PRA models also having design-basis functions during seismic events or functions credited for mitigation and prevention of severe accidents caused by seismic events 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 described in the SCD.

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

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

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

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

Based on the above, the Summary/Conclusion/Recommendation from Section 2.2.3 of Reference [4] applies to QCNPS, i.e., QCNPS is a Tier 1 plant for which the GMRS is very low such that unique seismic categorization insights are expected to be minimal. As discussed in Reference [4] the likelihood of identifying a unique seismic insight that would cause an SSC to be designated HSS is very low. Calvert Cliff's references [24], [25], and [26] provide additional supporting bases for Tier 1 plants and apply to QCNPS. Therefore, with little to no anticipated unique seismic insights, the 10 CFR 50.69 categorization process using the FPIE PRA and other risk evaluations along with the defense-in-depth and qualitative assessment by the IDP adequately identify the safety-significant functions and SSCs.

3.2.4 Other External Hazards

All external hazards, except for seismic, were screened for applicability to QCNPS using a plant-specific evaluation in accordance with Generic Letter 88-20 (Reference [27]) and updated to use the criteria in ASME PRA Standard RA-Sa-2009. Attachment 4 of this enclosure 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 and Shutdown

Consistent with NEI 00-04, the QCNPS categorization process will use the shutdown safety management plan described in NUMARC 91-06 (Reference [7]) 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 CEG risk management process ensures that the PRA models used in this application continue to reflect the as-built and as-operated plant for QCNPS. 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, CEG will implement a process that addresses the requirements in NEI 00-04, Section 11, "Program Documentation and Change Control." The process will review the results of periodic and interim updates of the plant PRA that may affect the results of the categorization process. If the results are affected, adjustments will be made as necessary to the categorization or treatment processes to maintain the validity of the processes. In addition, any PRA model upgrades will be peer reviewed prior to implementing those changes in the PRA model used for categorization.

3.2.7 PRA Uncertainty Evaluations

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

Uncertainty evaluations associated with the risk categorization process are addressed using the processes discussed in Section 8 of NEI 00-04 and in the prescribed sensitivity studies discussed in Section 5.

In the overall risk sensitivity studies, CEG will utilize a factor of 3 to increase the unavailability or unreliability of LSS components consistent with that approved for Vogtle in Reference [5]. Consistent with the NEI 00-04 guidance, CEG 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 (Reference [28]) and Section 3.1.1 of EPRI TR-1016737 (Reference [29]). 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.

Each PRA element notebook was reviewed for assumptions and sources of uncertainties. The characterization of assumptions and sources of uncertainties are based on whether the assumption and/or source of uncertainty is key to the 50.69 application in accordance with RG 1.200 Revision 2. Key QCNPS 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 QCNPS 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 [30]), consistent with NRC Regulatory Information Summary (RIS) 2007-06 (Reference [31]).

Finding and Observation (F&O) closure reviews were conducted on the PRA models discussed in this section. 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 [32]) as accepted by NRC in the letter dated May 3, 2017 (Reference [33]). The results of this review have been documented in Attachment 3 as well as the cited references, which are available for NRC audit.

Full Power Internal Events and Internal Flooding (FPIE) PRA Model

The QCNPS FPIE PRA model was peer reviewed in April 2017 using the NEI 05-04 process (Reference [34]); the PRA Standard, ASME/ANS RA-Sa-2009 (Reference [35]); and RG 1.200, Revision 2. This Peer Review (Reference [36]) was a full-scope review of the technical elements of the Internal Events and Internal Flooding, at-power PRA.

The FPIE PRA Peer Review team determined there were 31 unique finding level F&Os resulting in 14 Not-Met SRs. The findings from the Peer Review have been addressed in the Internal Events PRA model.

In February/March 2021, an F&O Closure Review (Reference [37]) was conducted for QCNPS. The scope of the review included the Internal Events and Internal Flooding PRA model. The F&O Independent Assessment Team closed all finding level F&Os. Currently, there are no open finding level F&Os against the FPIE PRA model.

Given there are no partially resolved or open findings, the QCNPS FPIE PRA is of adequate technical capability to support the 10 CFR 50.69 program.

Fire PRA Model

The QCNPS Fire PRA (FPRA) Peer Review (Reference [38]) was performed in June 2013 using the NEI 07-12 Fire PRA peer review process (Reference [39]); the ASME PRA Standard, ASME/ANS RA-Sa-2009 (Reference [35]); and RG 1.200, Revision 2 (Reference [30]). The purpose of this review was to establish the technical adequacy of the FPRA for the spectrum of potential risk-informed plant licensing applications for which the FPRA may be used. The FPRA Peer Review was a full-scope review of all technical elements of the QCNPS at-power FPRA against all technical elements in Part 4 of the ASME/ANS PRA Standard, including the referenced Internal Events Supporting Requirements (SRs) in Part 2. The Fire PRA Peer Review team determined there were 83 unique finding level F&Os resulting in 49 Not-Met SRs.

In February/March 2021, an F&O Closure by Independent Assessment was conducted for QCNPS (Reference [37]). The scope of the review included all FPRA Peer Review finding level F&Os from the 2013 peer review; 76 of the 2013 peer review finding level F&Os were closed during this F&O closure review. Of the 49 SRs that were previously assessed as Not-Met, 47 were assessed as Met to at least Capability Category II and two remained as Not-Met.

A FPRA focused scope peer review (FSPR) was conducted in February 2021 (Reference [40]). This FSPR resulted in an additional five finding level F&Os. All reviewed SRs were Met to at least a Capability Category II.

In May 2021, a follow-on F&O closure review was conducted to assess the remaining seven finding level F&Os from the February 2021 F&O closure review and the five finding level F&Os from the February 2021 FSPR (Reference [41]). All 12 finding level F&Os were assessed as closed during this review and all reviewed SRs were Met to at least a Capability Category II (including the two SRs that previously remained Not-Met). Therefore, all previous finding level F&Os were closed, and all applicable SRs were assessed as Met to at least Capability Category II.

Finally, another FSPR was conducted, also in May 2021 (Reference [42]). This FSPR resulted in one documentation finding level F&O (F&O 9-1) and one documentation SR assessed as Not-Met (SR FQ-F1). The F&O identified that detailed descriptions of significant contributors including cutsets, accident sequences, etc., were not provided. This F&O is addressed by and resolved in the FPRA documentation prepared for the model used in support of this application. Detailed descriptions of significant cutsets, accident sequences, etc. were added to the documentation; however, a formal F&O Closure review has not been completed for this remaining finding.

Given there are no partially resolved or open technical findings that may impact the 10 CFR 50.69 program, and the only remaining finding is related to documentation, it is concluded that the QCNPS FPRA is of acceptable technical capability to support the 10 CFR 50.69 program.

Conclusion

The above discussion demonstrates that the QCNPS PRA models are of sufficient quality and level of detail to support the categorization process. Additionally, it is concluded that each model has been subjected to a peer review process assessed against a standard or set of acceptance criteria that is endorsed by the NRC as required by 10 CFR 50.69(c)(1)(i). Attachment 3 summarizes the results of this effort.

3.4 Risk Evaluations (10 CFR 50.69(b)(2)(iv))

The QCNPS 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 process (i.e., performance monitoring) as it pertains to the proposed QCNPS Tier 1 approach discussed in Section 3.2.3, implementation of the CEG 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 CEG's 10 CFR 50.69 program documents. The program requires that the periodic review assess changes that could impact the categorization results and provides the Integrated Decision-making Panel (IDP) with an opportunity to recommend categorization and treatment adjustments. Station personnel from engineering, operations, risk management, regulatory affairs, and others have responsibilities for preparing and conducting various performance monitoring tasks that feed into this process. The intent of the performance monitoring reviews is to discover trends in component reliability; to help catch and reverse negative performance trends and take corrective action, if necessary.

The CEG configuration control process ensures that changes to the plant, including a physical change to the plant and/or 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 implementation. 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.

CEG 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 CEG 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 upgraded, a review of the SSC categorization will be performed.

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

4 REGULATORY EVALUATION

4.1 Applicable Regulatory Requirements/Criteria

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

- 10 CFR 50.69, "Risk-Informed Categorization and Treatment of Structures, Systems and Components for Nuclear Power Reactors"
- Regulatory Guide 1.201, "Guidelines for Categorizing Structures, Systems, and Components in Nuclear Power Plants According to their Safety Significance," Revision 1, May 2006 (Reference [3]).
- 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, May 2011 (Reference [43]).
- Regulatory Guide 1.200, "An Approach for Determining the Technical Adequacy of Probabilistic Risk Assessment Results for Risk-Informed Activities," Revision 2, March 2009 (Reference [30]).

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

4.2 No Significant Hazards Consideration Analysis

In accordance with 10 CFR 50.90, "Application for amendment of license, construction permit, or early site permit," Constellation Energy Generation, LLC (CEG) requests an amendment to Renewed Facility Operating License Nos. DPR-29 and DPR-30 for Quad Cities Nuclear Power Station (QCNPS), Units 1 and 2, respectively. Specifically, CEG proposes to modify the QCNPS licensing basis to allow for the voluntary implementation of the provisions of 10 CFR 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 either will not be changed or will be enhanced. This allows improved focus on equipment that has high safety significance resulting in improved plant safety.

According to 10 CFR 50.92, "Issuance of amendment," paragraph (c), a proposed amendment to an operating license involves no significant hazards consideration if operation of the facility in accordance with the proposed amendment would not:

(1) Involve a significant increase in the probability or consequences of an accident previously evaluated; or
(2) Create the possibility of a new or different kind of accident from any accident previously evaluated; or
(3) Involve a significant reduction in a margin of safety.

CEG has evaluated the proposed change for QCNPS by using the criteria in 10 CFR 50.92 and has determined that the change does not involve a significant hazards consideration. The following information is provided to support a finding of no significant hazards consideration.

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, CEG 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

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

CEG has established procedure(s) for use within the fleet for the use of the categorization process on a plant system. These CEG fleet procedures will be implemented at QCNPS prior to the use of the categorization process at QCNPS. The fleet procedures to be implemented at QCNPS 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 of the enclosure). 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, Revision 2.

- 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

Units	Model	Baseline CDF	Baseline LERF	Comments
Full Power Internal Events &Internal Flooding PRA Model				
1 and 2	QC118AD Model (Unit 1) QC118AD Model (Unit 2)	3.9E-6/yr (Unit 1) 3.9E-6/yr (Unit 2)	2.4E-7/yr (Unit 1) 2.4E-7/yr (Unit 2)	2021 FPIE Application Specific Model (ASM)
Fire Model				
1 and 2	QC118AFF1 Model (Unit 1) QC218AFF1 Model (Unit 2)	3.8E-5/yr (Unit 1) 4.3E-5/yr (Unit 2)	3.2E-6/yr (Unit 1) 3.5E-6/yr (Unit 2)	2021 Fire PRA Application Specific Model (ASM)

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

There are no partially resolved or open peer review findings or self-assessment open items for the QCNPS Internal Events and Internal Flooding PRA Model.

There are no partially resolved or open peer review findings or self-assessment open items for the QCNPS Fire PRA (FPRA) Model. There is one open finding related only to documentation, specifically that detailed descriptions of significant contributors including cutsets, accident sequences, etc., were not provided. This F&O is addressed by and resolved in the FPRA documentation prepared for the model used in support of this license amendment request (LAR). Detailed descriptions of significant cutsets, accident sequences, etc. were added to the documentation; however, a formal F&O closure review has not been completed for this remaining finding.

Attachment 4: External Hazards Screening

External Hazard	Screening Result		
	Screened? (Y/N)	**Screening Criterion (Note a)**	**Comment**
Aircraft Impact	Y	PS2 PS4	Acceptance criterion 1.A of Standard Review Plan 3.5.1.6 (Reference [44]) states the probability is considered to be less than an order of magnitude of 10^{-7} per year by inspection if the plant-to-airport distance D is between 5 and 10 statute miles, and the projected annual number of operations is less than 500 D^2, or the plant-to-airport distance D is greater than 10 statute miles, and the projected annual number of operations is less than 1000 D^2 **(PS2, PS4)**. Per the IPEEE (Reference [45]), the closest airport to the plant is the Clinton Municipal Airport, a small, public, general aviation facility located approximately 9 miles north-northwest of the plant. Airport data (Reference [46]) shows the annual options from this airport is less than 12,000 which is less than 500 D^2 **(PS2, PS4).** Quad Cities International Airport, about 21 miles southwest of the plant, is the nearest airport with scheduled commercial air service. Airport data (Reference [47]) shows the annual operations from this airport to be less than 29,000 which is less than the 1000 D^2 criteria **(PS2, PS4)**. Based on this review, the aircraft impact hazard is considered negligible.
Avalanche	Y	C3	The mid-western location of the plant precludes the possibility of an avalanche. Based on this review, the avalanche hazard can be considered negligible.
Biological Event	Y	C1	Per UFSAR Section 9.2.1.4 (Reference [48]), piping and heat exchanger intrusion by bivalves (such as Asiatic Clams and Zebra Mussels) has been identified as a potential hazard to the QCNPS RHR service water and diesel generator cooling water systems. QCNPS has implemented a program to trend

<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>flow blockage characteristics. This information is used to ensure flow blockage will not occur in safety-related systems using river water. The program includes:

• Periodic inspection of the intake bays.
• Periodic flushing of infrequently used or stagnant lines in safety-related service water systems.
• Annual water and substrate sampling.
• Periodic testing, inspection and cleaning of safety-related heat exchangers.
• Periodic inspection of high-flow and low-flow service water piping for corrosion, erosion, silting and biofouling.
• A long-term program for ultrasonic test examination of cooling water lines associated with safety-related heat exchangers.
• Corrosion coupons installed in DGCW and RHRSW piping.

Based on this review, the biological events hazard can be considered negligible.</td></tr>
<tr><td>Coastal Erosion</td><td>Y</td><td>C1

C5</td><td>Coastal erosion is a slowly developing event and could be mitigated or adequately responded to (C5).

Also, per UFSAR Section 2.4.4 (Reference [48]) the entire intake flume from the crib house to the river's edge is stripped out to the natural rock which is approximately elevation 557 feet. The natural river bottom between the river's edge and the main river channel varies in elevation from elevation 557 feet to elevation 565 feet, thus preventing a direct flow of water from the main channel to the crib house during a broken dam condition.

UFSAR Section 3.7.3.3.3 states that the retaining wall was analyzed for both an OBE and a DBE. The resulting stresses were below the allowables for both cases.</td></tr>
</table>

External Hazard	Screening Result		
	Screened? (Y/N)	Screening Criterion (Note a)	Comment
			UFSAR Section 3.7.3.3.4 states that the earth embankment was found to be capable of resisting the sliding effects during an SSE for all three cases considered. The cases were (1) circular sliding surface; (2) plane sliding surface; and (3) block sliding horizontally (**C1**). Based on this review, the coastal erosion hazard can be considered negligible.
Drought	Y	C5	Drought is a slowly developing hazard allowing time for orderly plant reductions, including shutdowns. Based on this review, the drought hazard can be considered negligible.
External Flooding	Y	C1 PS4	The external flooding evaluation was performed and documented in the Flood Hazard Reevaluation Report (Reference [49]) and the Integrated Assessment (IA) Report (Reference [50]). The NRC confirmed in its review of the IA (Reference [51]) that QCNPS has appropriately addressed plant vulnerabilities to external flooding and will not require additional safety enhancements since mitigating strategies (FLEX) remain feasible (per Reference [52]) for the reevaluated flood hazard and effective protection of Key Safety Functions (KSFs) have been demonstrated for higher likelihood flooding scenarios. There are only two external flood causing mechanisms that pose a challenge to the site that required additional screening: 1) Local Intense Precipitation (LIP) and 2) the combination of PMF, upstream hydrologic dam failure, and coincident wind/wave action. The LIP mechanism does not pose a challenge to the plant once six temporary barriers are installed around the power block reliably preventing water from entering the buildings. Table A4-1 includes a list of LIP flood barriers credited for screening this hazard. (**C1**)

External Hazard	Screening Result		
	Screened? (Y/N)	**Screening Criterion (Note a)**	**Comment**
			For the combined effects floods, a probabilistic flood hazard analysis was performed to characterize the frequency of floods exceeding plant grade and the LIP barriers. Refer to Figure A4-2. Following minor revisions to QCOA 0010-16 (Reference [53]), the LIP barriers will be installed prior to the arrival of flood waters from the combined event protecting the plant up to an elevation of 599.0’ which equates to a frequency of approximately 5E-8/yr and is considered Scenario 2. The frequency of water exceeding plant grade of 595.0’ is conservatively assumed to be 2E-6/yr and is considered Scenario 1. The frequency of floods exceeding 595.0' but less than 599.0' is defined as the difference between Scenario 1 and 2 frequencies and is 1.95E-6/yr. A conservative analysis of CDF was performed (Reference [54]). The analysis includes the assumption that failure to install the LIP gates and for them to not perform their designed function fails all flood mitigation and leads directly to core damage. All other mitigation capabilities, such as the CLB strategy/FLEX, temporary dams, and that the plant will be in a shutdown condition, are conservatively ignored. The installation and failure of the six barriers that require manual actions is conservatively assumed to have a failure probability of 0.3. Given the assumption of no additional mitigation capabilities, the CCDP is very conservatively estimated to be equal to the failure of the LIP gates (0.3). When combining the CCDP with the frequency for the two scenarios of flood waters exceeding plant grade and then topping the LIP barriers, the bounding mean CDF can be estimated at 2.5E-7/yr. (**PS4**) Based on this review, the risk from external flood hazard can be considered negligible and screened from further evaluation.

External Hazard	Screening Result		
	Screened? (Y/N)	Screening Criterion (Note a)	Comment
Extreme Wind or Tornado	Y	PS4	Section 5.2.1 of the IPEEE (Reference [45]) documents the screening of high winds and tornadoes; the screening is based on design information from UFSAR Section 3.3 (Reference [48]). Section 3.3.2.1 of the IPEEE provides the design parameters for tornado loadings, which include 300 mph tangential velocity, 60 mph translational velocity and a pressure drop of 3 psi in 3 seconds. Tornado wind speeds and design pressures bound straight winds for most plant structures, including the exterior concrete walls of the reactor building, turbine building, control room, and steel superstructure of the reactor building and turbine building. However, per Section 3.3.1.1.2 of the UFSAR, the concrete chimney can only withstand wind pressures up to 217 mph. Tornado wind speed hazard curve information for QCNPS is provided in Table 6-1 of NUREG/CR-4461, Revision 2 (Reference [55]). Based on the Enhanced Fujita (EF)-scale, the wind speed for the 1E-6 annual exceedance probability (AEP) is 202 mph. Comparable 1,000,000 year Mean Recurrence Interval (MRI) tornado wind speeds from the ASCE 7 Hazard Tool (Reference [56]) are less than 200 mph. Therefore, the frequency of the design tornado wind speed for Class I structures is much less than 1E-6/yr and the frequency of winds that could cause the failure of the concrete chimney are also less than 1E-6/yr. Thus, wind pressure effects can be screened using Criterion PS4. Although QCNPS conforms to its design basis and licensing basis for tornado missiles, there are several risk significant SSCs (e.g., DG intake and exhaust pipes, certain safety-related switchgear, and various electrical cables) that are not protected from all tornado missiles. Because of this, conservative tornado missile risk models were developed for both units. The models use the ASCE 7 Hazard Tool data to determine the hazard curve along with conservative failure probabilities and assumptions (e.g., loss of offsite power with no recovery,

External Hazard	Screening Result		
	Screened? (Y/N)	**Screening Criterion (Note a)**	**Comment**
			guaranteed failure of the SBO DGs). Estimates for CDF for both units were less than 1E-6/yr and LERF was estimated to be much less than 1E-7/yr. Therefore, the tornado missile hazard can also be screened using Criterion PS4.
Fog	Y	C4	The principal effects of such events (such as freezing fog) would be to cause a loss of off-site power, which is addressed in weather-related LOOP scenarios in the FPIE PRA model for QCNPS. Based on this review, the fog hazard can be considered negligible.
Forest or Range Fire	Y	C3 C4	Per IPEEE Table 5-1 (Reference [45]), the site landscaping and lack of forestation prevent such fires from posing a threat to QCNPS (**C3**). In addition, forest fires originating from outside the plant boundary may cause a loss of offsite power event, which is addressed for grid-related LOOP scenarios in the FPIE PRA model for QCNPS (**C4**). Based on this review, the forest fire hazard can be considered negligible.
Frost	Y	C4	The principal effects of such events would be to cause a loss of off-site power, which is addressed for weather-related LOOP scenarios in the FPIE PRA model for QCNPS. Based on this review, the frost hazard can be considered negligible.
Hail	Y	C1 C4	The principal effects of such events would be to cause a loss of off-site power, which is addressed for weather-related LOOP scenarios in the FPIE PRA model for QCNPS (**C4**). Flooding impacts are covered under External Flooding/Intense Precipitation (**C1**).

External Hazard	Screening Result		
	Screened? (Y/N)	**Screening Criterion (Note a)**	**Comment**
			Based on this review, the hail hazard can be considered negligible.
High Summer Temperature	Y	C1 C4 C5	Per UFSAR Section 1.2.1 (Reference [48]), the design of components important to safety of the units and the station includes allowances for environmental phenomena (**C1**). Per UFSAR Section 9.2.5.3 (Reference [48]), recirculation of the ultimate heat sink (UHS) volume would result in elevated RHRSW and DGCW temperatures. The worst-case scenario would be a dam failure during the summer months with a river temperature of 95°F and minimum evaporative cooling. Assuming a conservative UHS volume of 2.15 million gallons of water, 5100 gpm of makeup from the portable pumps and 24 hours of shutdown time on the main condenser, this results in a maximum RHRSW and DGCW inlet temperature of 109°F. This would be sufficient for each unit to operate one RHR pump, one RHR heat exchanger, one RHRSW pump and one DGCW pump for a total flow of approximately 10,000 gpm for cooling (**C1**). In addition, continuous use procedure QCOP 0010-10, "Required Hot Weather Inspections" (Reference [57]) contains guidance for inspections of various plant locations and equipment during hot weather conditions (**C5**). 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 hazard can be considered negligible.

External Hazard	Screening Result		
	Screened? (Y/N)	Screening Criterion (Note a)	Comment
High Tide, Lake Level, or River Stage	Y	C3	Since the site is located on the Mississippi River, the high tide hazard does not apply. Based on this review, the high tide hazard can be considered negligible.
Hurricane	Y	C3	Hurricanes are extreme tropical storms that originate offshore and as such do not reach QCNPS due to the mid-western location of the site. Based on this review, the hurricane hazard can be considered negligible.
Ice Cover	Y	C1 C4	UFSAR Section 2.4.7 (Reference [48]) discusses ice cover. An ice-melting line is tied into the side of the discharge flume upstream of a weir with a top elevation at 574.75 feet. This line is a 96-inch diameter pipe with a bottom elevation at that point of 557 feet. A gate is provided in this line for shutoff. The ice-melting line runs into and across the intake flume and is provided with four outlets having a bottom elevation of 558 feet. During frozen conditions, the circulating water system would be in service, and the ice melting line would normally be opened enough to keep the intake forebay free of ice (**C1**). The principal effect of ice cover events would be to cause a loss of off-site power event, which is addressed for weather-related LOOP scenarios in the FPIE PRA model for QCNPS (**C4**). Based on this review, the ice cover hazard can be considered negligible.
Industrial or Military Facility Accident	Y	C1 C3	Per the IPEEE (Reference [45]), and UFSAR Section 2.2 (Reference [48]), there are no military facilities within a 5-mile radius of QCNPS (**C3**). UFSAR Section 2.2 further states that within a 5-mile radius of QCNPS, the general character of land use is rural, comprised of scattered villages and homes, except for two industrial areas.

External Hazard	Screening Result		
	Screened? (Y/N)	**Screening Criterion (Note a)**	**Comment**
			The industrial areas located within a 5-mile radius of include the Cordova Industrial Park, which has the nearest major industrial tenant, the Minnesota Mining and Manufacturing (3M) plant, a chemical company situated 1 ½ miles from the site. There is also the CF Industries Chemical complex, located 3.1 miles north of the site. Per UFSAR Section 2.2.2, none of the operations at Cordova Industrial Park pose an explosion, explosive shock, resulting missiles, or toxic fumes release threat to QCNPS. Furthermore, there is no chlorine gas used in the area (**C1, C3**). In addition, to mitigate the impact of an industrial facility accident, QCOA 0010-17, "Toxic Gas/Chemical Release from Nearby Facilities" (Reference [58]) includes automatic (control room ventilation system isolation) and subsequent operator actions (don air packs, obtain temporary ventilation, etc.). See also Toxic Gas. Based on this review, the industrial or military facility accident hazard can be considered negligible.
Internal Flooding	N/A	N/A	The QCNPS Internal Events PRA includes evaluation of risk from internal flooding events.
Internal Fire	N/A	N/A	The QCNPS Internal Fire PRA model addresses risk from internal fires.
Landslide	Y	C3	Plant site is located on level terrain and is not subject to landslides. Additionally, the mid-western location of QCNPS precludes the possibility of a landslide. Based on this review, the landslide hazard can be considered 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 QCNPS internal events model through the incorporation of generic and plant specific data. Based on this review, the lightning hazard can be considered negligible.
Low Lake Level or River Stage	Y	C1 C5	Per UFSAR Section 9.2.5.2 (Reference [48]), the UHS is provided to mitigate the consequences of the postulated failure of Lock and Dam No. 14 on the Mississippi River downstream of the plant, which would cause river level to drop. The station design is such that if Lock and Dam No. 14 were to fail, the water level would recede in the intake bay to the point where it would be separated from the river. UFSAR Section 9.2.5.2 states that the UHS is defined as the water that is captured inside the log boom located in the discharge canal. This captured volume is approximately 3 million gallons (**C1**). Use of the UHS to shut down the reactors requires the operation of portable diesel pumps along with RHRSW and DGCW to shutdown the units. Following a dam failure, approximately 2 days would be available to position the portable pumps (**C5**). Per UFSAR Section 2.4.4, if a natural disaster such as an earthquake were to occur, the dam would most likely not incur the complete loss of a function. Such a disaster would likely result in malfunctioning or loss of gate operating capability. This condition would not result in the loss of the pool used as the plant's UHS but simply the inability to operate the Lock & Dam gates. Therefore, the maximum credible failure is considered the loss of both the upstream and downstream lock miter gates (**C1**).

External Hazard	Screening Result		
	Screened? (Y/N)	**Screening Criterion (Note a)**	**Comment**
			Continuous use procedure QCOA-001014, "Lock and Dam #14 Failure" (Reference [59]) also provides direction for loss of river due to dam failure. In the procedure, use of the ultimate heat sink requires taking in river water through the discharge piping and returning it over the log boom at the intake flume. Based on this review, the low lake or river water level hazard can be considered negligible.
Low Winter Temperature	Y	C1 C4 C5	Per UFSAR Section 1.2.1 (Reference [48]), the design of components important to safety of the units and the station includes allowances for environmental phenomena (**C1**). In addition, continuous use procedure QCOP 0010-02, "Required Cold Weather Inspections" (Reference [60]) contains guidance for inspections of various plant locations and equipment during cold weather conditions (**C5**). 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**). See also Ice Cover. Based on this review, the low winter temperature hazard can be considered negligible.
Meteorite or Satellite Impact	Y	PS4	Per the IPEEE (Reference [45]), the frequency of a meteor or satellite strike is judged to be so low as to make the risk impact from such events insignificant. Based on this review, the meteorite or satellite hazard can be considered negligible.

External Hazard	Screening Result		
	Screened? (Y/N)	**Screening Criterion (Note a)**	**Comment**
Pipeline Accident	Y	C1 C3	Per IPEEE Table 5-1(Reference [45]), the nearest pipelines are one mile from the plant and consist of a 3-inch diameter and a 12-inch diameter pipe. Both are natural gas pipelines. There is no significant hazard to the plant from these lines. Other lines, located three miles or more from the site, carry ethane, ethylene, and propane through 12-inch pipes. These lines have been evaluated in a "Control Room Habitability Study for QCNPS Units 1 and 2" as cited in Section 2.2 of the UFSAR (Reference [48]). No significant toxic hazard exists from pipelines near the site (**C1, C3**). See also Industrial or Military Facility Accident, Toxic Gas, and Transportation Accidents. Based on this review, the pipeline accident hazard can be considered negligible.
Precipitation, Intense	Y	C1	The Integrated Assessment (Reference [50]) confirmed that the site has provided adequate protection from the LIP flooding mechanism. Thirteen LIP barriers are credited for screening this hazard and thus will be considered high safety significant (HSS) should their associated systems be categorized (Table A4-1)). See also External Flooding. Based on this review, the risk from the intense precipitation hazard can be considered negligible.
Release of Chemicals in Onsite Storage	Y	C1	Per IPEEE Table 5-1(Reference [45]), chlorination of water systems is performed using a hypochlorite system. No chlorine gas is stored on-site. Various acids and caustics are stored on-site but pose no hazard to the plant. Hydrogen and liquid oxygen are stored on-site but compliance with EPRI guidelines ensures that no significant hazard results from these materials.

External Hazard	Screening Result		
	Screened? (Y/N)	Screening Criterion (Note a)	Comment
			Per UFSAR Sections 6.4.4.2.1- 6.4.4.2.2 (Reference [48]), models were developed to calculate the concentrations of toxic chemicals in the control room in the event of an accidental spill consistent with the models described in NUREG-0570 (Reference [61]). Based on the physical and toxicological properties of the chemicals stored at the QCNPS site, it is concluded that none of the chemicals are of concern. For these chemicals, the unisolated control room concentrations will not exceed the threshold limit value in the event of a postulated release. See also Toxic Gas. Based on this review, the release of chemicals from onsite storage hazard can be considered negligible.
River Diversion	Y	C1	Per UFSAR 2.4.9 (Reference [48]), the authority to control the river is vested in the U.S. Army Corps of Engineers. Should the need to control the river arise, CEG will make the required notification to the Corps of Engineers. These arrangements have been made in advance and described in the QCNPS emergency procedures. The NRC confirmed in its evaluation of the QCNPS Flood Hazard Reevaluation Report that flooding from channel migrations or diversions is not a plausible flood hazard mechanism at the QCNPS site (Reference [62]). Based on this review, the river diversion hazard can be considered negligible.
Sand or Dust Storm	Y	C1	Per the IPEEE (Reference [45]), the mid-western location of QCNPS prevents sandstorms. More common wind-borne dirt can occur but poses no significant risk given the robust structures and protective features of the plant.

External Hazard	Screening Result		
	Screened? (Y/N)	Screening Criterion (Note a)	Comment
			Based on this review, the sandstorm hazard can be considered negligible.
Seiche	Y	C3	Per UFSAR Section 2.4.5 (Reference [48]), flooding due to seiches is not applicable to QCNPS Station. In its review of the QCNPS Flood Hazard Reevaluation Report (Reference [52]), the NRC confirmed that flooding due to seiche is not a plausible flooding mechanism due to the location of the site in relation to large bodies of water, and therefore does not impact the site. Based on this review, the seiche hazard can be considered negligible.
Seismic Activity	N/A	N/A	See Section 3.2.3 of the enclosure and Figure A4-1 in this attachment.
Snow	Y	C4 C5	This hazard is slow to develop and can be identified via monitoring and managed via normal plant processes (**C5**). Potential flooding impacts are accounted for under external flooding screening (**C4**). Based on this review, the snow hazard can be considered negligible.
Soil Shrink-Swell Consolidation	Y	C1 C5	The potential for this hazard is low at the site, the plant design considers this hazard (**C1**), and the hazard is slow to develop and can be mitigated (**C5**). Based on this review, the soil shrink-swell consolidation impact hazard can be considered negligible.
Storm Surge	Y	C3	Per UFSAR Section 2.4.5 (Reference [48]), flooding due to surges is not applicable to QCNPS Station.

External Hazard	Screening Result		
	Screened? (Y/N)	Screening Criterion (Note a)	Comment
			In its review of the QCNPS Flood Hazard Reevaluation Report (Reference [52]), the NRC confirmed the location of the site in relation to large nearby waterbodies and the site is approximately 140 miles from the coast of Lake Michigan, which would be only source of a storm surge, and the topography between the site and Lake Michigan would attenuate a storm surge and the associated wave runup (Reference [62]). The NRC confirmed the licensee's conclusion that storm surge is not a plausible flood hazard mechanism at QCNPS and would not impact the site. Based on this review, the storm surge hazard can be considered negligible.
Toxic Gas	Y	C1	Per UFSAR Section 2.2.3.2 (Reference [48]), liquid hydrogen and liquid oxygen storage facilities are installed at the site. Compliance with EPRI guidelines (Reference [63]) ensures that the system installation and operation will not produce a safety concern. Additionally, the onsite delivery routes for transporting hydrogen and oxygen to their respective storage facilities comply with EPRI guidelines and therefore will not produce a safety concern. UFSAR Section 6.4.4.2 (Reference [48]) outlines the toxic gas analysis. Based on the physical and toxicological properties of the chemicals stored at the QCNPS site, it is concluded that none of the chemicals are of concern. For these chemicals, the unisolated control room concentrations will not exceed the threshold limit value (TLV) in the event of a postulated release. Per UFSAR Section 6.4.4.2.3, the control room HVAC system provides toxic gas protection to the control room emergency zone in case of either an onsite or offsite toxic chemical accident. The system provides this protection by either manual isolation through operator action or automatic isolation via a toxic gas analyzer.

External Hazard	Screening Result		
	Screened? (Y/N)	Screening Criterion (Note a)	Comment
			For ammonia, a monitor is provided since the control room concentrations reach toxicity limits faster than the operator can manually isolate the system after detection of odor. See also Transportation Accidents and Release of Chemicals from Onsite Storage. Based on this review, the toxic gas hazard can be considered negligible.
Transportation Accident	Y	C1 C3 C4	UFSAR Section 2.2 (Reference [48]) states that there are six transportation routes within a 5-mile radius of the plant: the Mississippi River, U.S. Route 67, State Route 84, and three railroad lines. Per the IPEEE (Reference [45]), rail, barge, aircraft, and pipeline transportation accidents are insignificant hazards to the site. (**C1, C3**) In addition, per UFSAR Section 6.4.4.2.3, the control room HVAC system provides toxic gas protection to the control room emergency zone in case of either an onsite or offsite toxic chemical accident. The system provides this protection by either manual isolation through operator action or automatic isolation via a toxic gas analyzer (**C1**). Other releases of toxic gases during transportation to local facilities are included in the toxic gas hazard screening (**C1, C4**). See also Toxic Gas. Based on this review, the transportation accidents hazard can be considered negligible.

External Hazard	Screening Result		
	Screened? (Y/N)	Screening Criterion (Note a)	Comment
Tsunami	Y	C3	Per UFSAR Section 2.4.6 (Reference [48]), flooding due to tsunamis is not applicable to QCNPS. In its review of the QCNPS Flood Hazard Reevaluation Report (Reference [52]), the NRC confirmed flooding due to tsunamis is not applicable to QCNPS. Based on this review, the tsunami hazard can be considered negligible.
Turbine-Generated Missiles	Y	PS3	For main turbine missiles, the NRC-preferred option for unfavorably oriented main turbines such as QCNPS (i.e., main turbine orientation such that essential SSCs are inside the RG 1.115 (Reference [64]) defined turbine missile strike zone), is to limit P_1, the annual probability of "low trajectory" turbine missile generation resulting in the ejection of turbine disk (or internal structure) fragments through the turbine casing, to 1E-5. A 2019 evaluation (performed consistent with RG 1.115 approaches) was performed to calculate the P_1 value for various cases of test intervals of turbine overspeed protection (OPS) components for the QCNPS main turbine (Reference [65]). The value of P_1 for QCNPS is the sum of the annual probabilities of the two RG 1.115 missile generation scenarios: 1) brittle fracture probability at design speed and 2) ductile failure probability given turbine overspeed which increases when main turbine overspeed protection system components are tested less frequently. Based on the monthly main turbine surveillance tests corresponding to the "Case 3 Test Interval" from Table 6-3 of Reference [65], the total P_1 probability is 8.01E-6. Per Table 1 of RG 1.115, the likelihood of main turbine missile impact on essential equipment is 1E-7/yr for a plant with an unfavorably oriented main

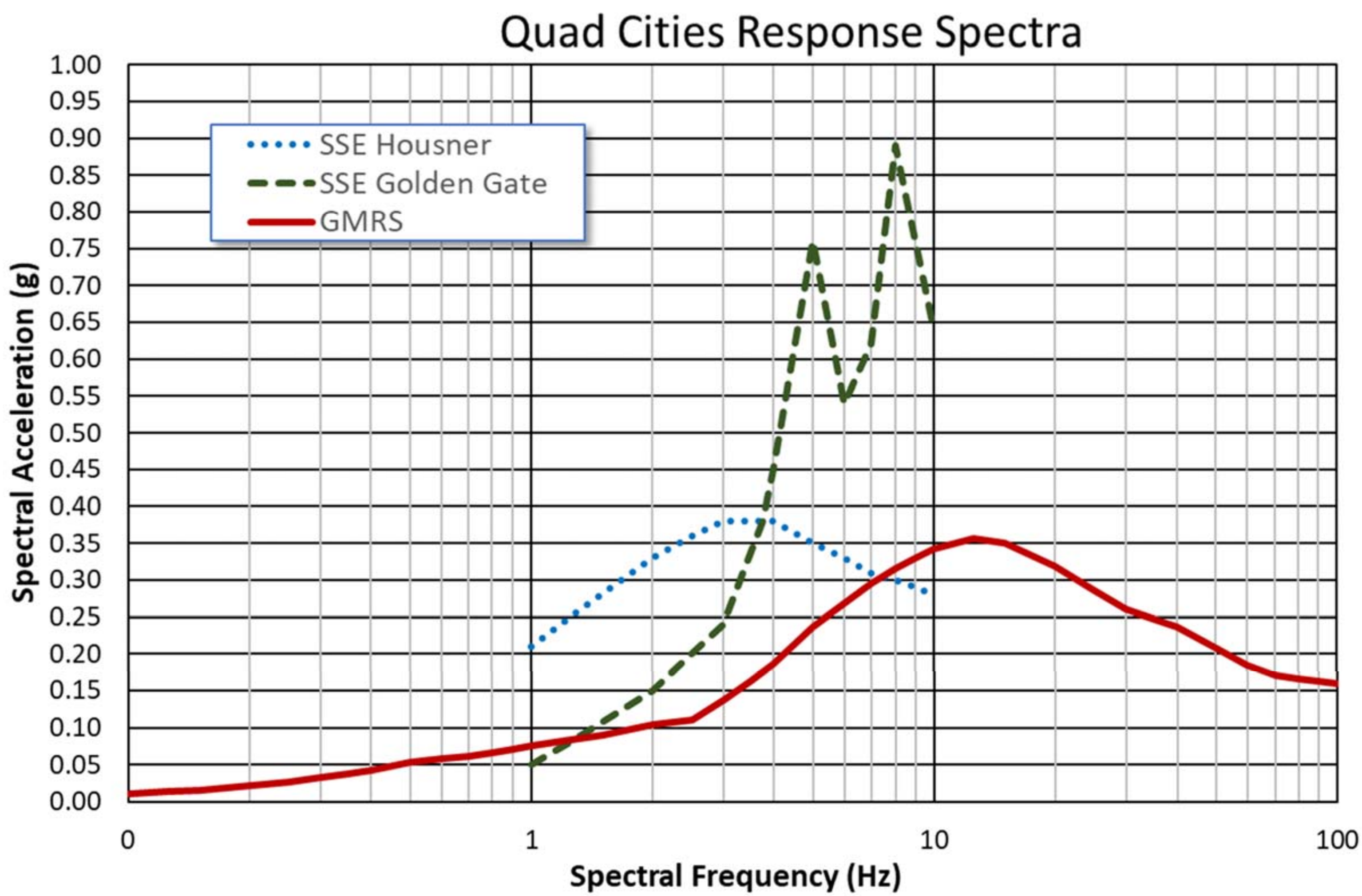
External Hazard	Screening Result		
	Screened? (Y/N)	Screening Criterion (Note a)	Comment
			turbine demonstrating a P_1 value of less than 1E-5/yr. As such, the estimated CDF from a QCNPS postulated main turbine missile is less than 1E-7/yr. Based on this review, the turbine missile hazard can be considered negligible.
Volcanic Activity	Y	C3	This hazard is not applicable to the site because of location (no active or dormant volcanoes located near the plant site). Based on this review, the volcanic activity hazard can be considered negligible.
Waves	Y	PS4	Wind-generated waves are subsumed in the combined effects flood mechanism which was screened with a CDF well below 1E-6/yr. See External Flooding. Based on this review, the waves hazard can be considered negligible.

Note a – See Attachment 5 for descriptions of the screening criteria.

Table A4-1: Barriers Credited for Screening External Flood Hazard

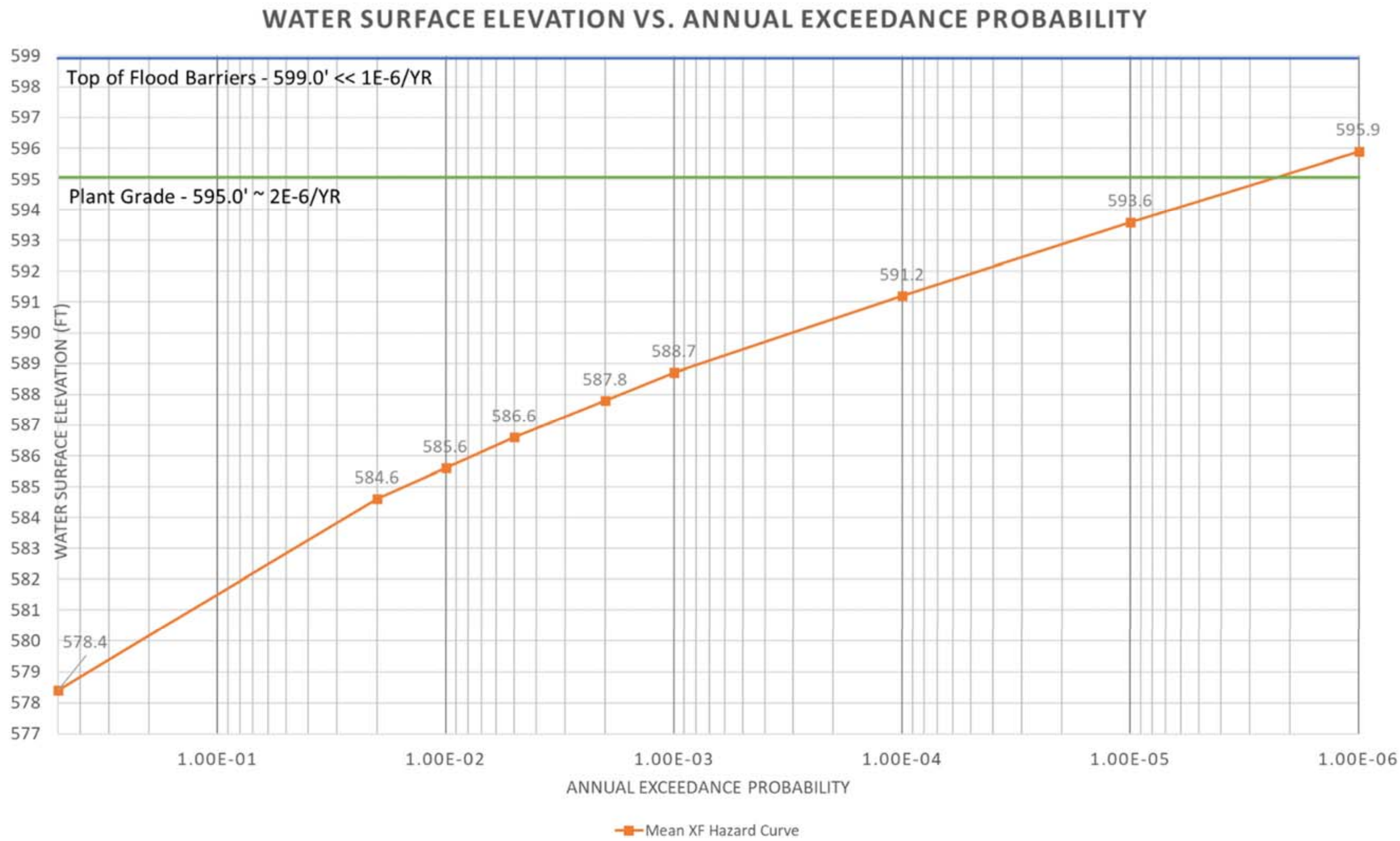
Barrier Number	Barrier Description	Requires Manual Closure
1A/1B	U1 & U2 Turbine Building Roll Up Doors (2 Barriers)	Yes
2A/2B	U1 & U2 Turbine Building/Reactor Building Interlock to HRSS Personnel Doors	No
3	Reactor Building 1/2 Trackway Door	No
4	1/2 EDG Interlock Door	No
5	Steel Plate on North Side of 1/2 EDG Building	No
6	Unit 2 Turbine Building Northwest Personnel Access Door	No
7	Turbine Building to Radwaste Building Door	No
8	LTD Building to Trackway 1 Door	Yes
9	Personnel Decon Room Door	Yes
10	Service Building to Trackway 1 Door	Yes
11	Aux Electric Room Door	Yes
TB North Siding	Turbine Building North Siding	No

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



(From Reference [15], Table 2.4-1 (GMRS), Table 3.1-1 (Golden Gate SSE)
and Table 3.1-2 (Housner SSE)

Figure A4-2: External Flooding Hazard Curve for Combined Effects Flooding



Attachment 5: Progressive Screening Approach for Addressing External Hazards

Event Analysis	Criterion	Source
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
	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
	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

The process defined in NUREG-1855 Revision 1 (Reference [28]) and Electric Power Research Institute (EPRI) Technical Reports 1016737 (Reference [29] and 1026511 (Reference [66]) was used to evaluate uncertainties in this application. These include:

- Identification of plant-specific Internal Events/Internal Flooding PRA model uncertainty sources, as well as generic sources per EPRI 1016737.
- Consideration of parameter and completeness uncertainties.
- Identification of plant-specific Internal Fire PRA model sources, along with generic sources per Appendix B of EPRI 1026511.
- Consideration of generic Level 2 model sources per Appendix E of EPRI 1026511, as applicable to Large Early Release Frequency (LERF).

The QCNPS Internal Events and Fire PRA models and documentation were reviewed for generic and plant-specific modeling assumptions and related sources of uncertainty. The applicable lists of EPRI-identified generic sources of uncertainty per EPRI 1016737 and EPRI 1026511 were also reviewed. Each PRA model includes an evaluation of the potential sources of uncertainty for the base case models using the approach that is consistent with the ASME/ANS RA-Sa-2009 (Reference [35]) requirements for identification and characterization of uncertainties and assumptions. The evaluations identify those sources of uncertainty that are important to the PRA results and may be important to PRA applications. The approach meets the intent of steps C-1 and E-1 of NUREG-1855, Revision 1 (Reference [28]).

These evaluations are documented for Internal Events and Internal Flooding in Appendix G of the Quantification Notebook, QC-PRA-014 (Reference [67]), for Internal Fire in the Uncertainty and Sensitivity Analysis Notebook, QC-PRA-021.62 (Reference [68]), and for Level 2 in the Level 2 Notebook, QC-PRA-015 (Reference [69]).

The results of the base PRA evaluations were reviewed to determine which potential uncertainties could impact applications, including the 50.69 Program. This evaluation meets the intent of the screening portion of steps C-2 and E-2 of NUREG-1855, Revision 1.

In addition, for the 50.69 Program, the guidance in NEI 00-04 (Reference [1]) specifies sensitivity studies to 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 [70] cites NUREG-1855, Revision 1, as related guidance. In Section B of RG 1.174, Revision 3, the guidance acknowledges specific revisions of NUREG-1855 to include changes associated with expanding the discussion of uncertainties. The results of the evaluation of PRA model sources of uncertainty are reflected in the Uncertainty Notebook for this application (Reference [71]).

> *Note*: As part of the required 50.69 PRA categorization sensitivity cases directed by NEI 00-04, Internal Events / Internal Flood and Fire PRA models, human failure events (HFEs) and common cause failure (CCF) basic events are increased to their 95th percentile and also decreased to their 5th percentile values. In addition, maintenance unavailability terms are set to 0.0. For the Fire PRA model only, a sensitivity case is required to allow no credit for manual suppression. These results can drive a component and respective functions to HSS and, therefore, the uncertainty of the

modeling of HFEs and CCF basic events in the PRA. The probabilities associated with those events (human error probabilities (HEPs) and CCF probabilities) are accounted for in the 50.69 Application.

Based on the above evaluations, key assumptions and sources of uncertainty which could affect this application were identified and dispositioned with sensitivity analyses using the QCNPS Internal Events and FPRA models as documented in Reference [71]. The results are summarized in the table below.

Sources of Assumption/ Uncertainty	10 CFR 50.69 Impact	Model Sensitivity and Disposition (10 CFR 50.69)
Internal Events and Internal Flood (IE/IF) Model		
Core Cooling Success Following Containment Failure or Venting Through Non-Hard Pipe Vent Paths Following containment failure, injection from Control Rod Drive (CRD) and Feedwater (FW)/Condensate could still be maintained, but if a large containment failure occurs, injection paths may be disrupted leading to loss of these external sources. This failure probability is based on a detailed structural analysis of the Mark I containment design and large scale ultimate failure testing of steel containments.	FW / Condensate / Steam Bypass Control System (SBCS) and CRD are credited for success after containment failure, but an additional basic event (1CNPVDWRUPT--R-- large Drywell (DW) containment failure causes loss of injection) is included that represents the likelihood that the containment failure size and location disrupts the capability of FW / Condensate / SBCS and CRD to inject.	A sensitivity analysis was performed that increased the conditional probability by a factor of 10 (to 6.0E-1) that a large drywell failure would result in loss of the feedwater, condensate, SBCS, and CRD injection capabilities. The results demonstrate that FPIE PRA and FPRA CDF and LERF results are sensitive to the failure probability associated with failure of all the selected injection systems following a large drywell failure. However, a factor of 10 increase in the conditional failure probability, applied to all of the selected injection systems, is not considered credible.
High Pressure Coolant Injection (HPCI) Room Cooling HPCI room cooling is supplied by DGCW1. It requires manual startup action. No room cooling is required for HPCI mission time as long as there is no gland seal condenser failure. For gland seal failures, the HPCI system is assigned failure directly.	Heating, Ventilation, and Air Conditioning (HVAC) dependencies for HPCI are not included for early operation but are included (fan only) for extended operation beyond 24 hours.	The requirement for room cooling for various HPCI mission times, with and without failure of the gland seal condenser, is identified as a candidate source of model uncertainty. A sensitivity analysis was performed that increased the failure probability for the HPCI gland seal hotwell pump failing to start by about a factor of 100 (to 1.0E-1). The results demonstrate that FPIE PRA CDF and LERF have little sensitivity to the failure probability associated with

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		failure of HPCI room cooling. However, a factor of about 100 increase in the pump's failure probability is conservative and not consistent with observed behavior.
Digital Feedwater Control Failure Probabilities There are model uncertainties associated with modeling digital systems, such as those related to determining the failure modes of these systems and components.	Reliability values from vendor studies demonstrating that the system performance would result in less than 0.1 transients per year are used for the key components of the system. Basic events representing the reliability values for the auto level controller, the field buses, false signal from the redundant reactivity control system, and false signal from the Level 8 trip system are included in the system logic model.	For this sensitivity analysis, the failure probability for the digital feedwater controller failing to control feedwater such that the Reactor Pressure Vessel (RPV) overfills and floods the steam line was increased by a factor of 100. The results demonstrate that FPIE PRA CDF and LERF are not sensitive to the failure probability associated with failure of the digital feedwater controller to stop feedwater prior to vessel overfill.
Instrument Air (IA) System Recovery (Containment Vent Valve Dependency on Air) The containment vent valves do not have accumulator backups to provide a method of successful venting given a loss of IA scenario. Currently, the model credits IA recovery at 24 hours to restore the hard pipe vent path.	It is assumed the containment vent valves cannot be opened by local manipulation of the valves or their air operators. They require instrument air to provide the force to open the containment vent AOVs. This requires instrument air availability to fulfill the containment vent function. For some low probability sequences, instrument air is not available due to system failures. However, the model uses data from NSAC-161 (Reference [72]) to provide a basis to support the restoration of instrument air or its support systems.	The recovery probability assigned for recovering instrument air is considered reasonable and is supported by data. However, use of this value could lead to a slightly optimistic assessment of containment vent success. For this sensitivity analysis, the probability of failing to recover balance of plant systems (including instrument air) after IA failure due to random causes was increased to 1.0. The results demonstrate that FPIE PRA CDF and LERF are relatively insensitive to the failure probability associated with failure to recover instrument air (balance

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		of plant, as well as specific to support of venting).
Diverse and Flexible Coping Strategies (FLEX) and Hardened Containment Vent System (HCVS) Human Error Probabilities (HEPs) and Equipment Failure Rates For the QCNPS IE PRA model there were no industry-approved data sources for FLEX equipment reliability. The QCNPS PRA models FLEX component failures and human failure events associated with failure to align the FLEX equipment.	The FLEX component failure rates can be represented by using the failure rates of "like-components" (e.g., emergency diesel generator (EDG)) as surrogates (e.g., for the FLEX diesel generators). The HEPs can be represented using screening values (ranging from 1E-3 to 0.5). FLEX component failures are estimated based on "like-components" by increasing that like-component's failure rate by a factor of two. Human error probabilities employ screening values.	The HEPs and equipment reliabilities used for FLEX may be underestimated given the current state of knowledge about FLEX. For this sensitivity case, credit for FLEX was completely removed from the model through use of a FLAG event. The results demonstrate that FPIE PRA CDF and LERF essentially have no sensitivity to the availability of FLEX. This result is driven by the fact that the use of FLEX at the QCNPS is constrained to situations declared as Extended Loss of AC Power (ELAP) scenarios (including station blackout (SBO)). These scenarios are rare, and thus there are few sequences in the PRA for which FLEX is credited. Removing that credit therefore has very little impact on the quantified results.
BlackStarTech (BST) portable carts equipment and human action reliability For the QCNPS PRA model there were no industry-approved data sources for BlackStarTech (BST) equipment reliability. The QCNPS PRA models BST component failures and human failure events associated with failure to align the BST equipment.	Use of BST has the same constraints as FLEX and HCVS (i.e., ELAP). The BST component failure rates can be represented by using the failure rates of "like-components" (e.g., batteries and battery chargers) as surrogates for BST equipment. The HEPs can be represented using standard HRA techniques (as employed for other human failure events throughout the PRA).	The base model includes credit for the BlackStarTech (BST) portable equipment (which includes the capability to supply specific AC and DC power to select components). Equipment failure probabilities are based on similar components (e.g., batteries), and human error probabilities are calculated using detailed human reliability analysis techniques.

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		For this sensitivity case, credit for BST was completely removed from the model through use of a FLAG event. The results demonstrate that FPIE PRA CDF and LERF have no sensitivity to the availability of BST. Use of BST has the same constraints as FLEX and HCVS (i.e., ELAP).
Fire Model		
Fire PRA component selection involves the selection of components to be treated in the analysis in the context of fire 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 Fire PRA assumes that, at a minimum, a plant trip occurs. This is consistent with accepted industry practice. The Fire PRA does not credit some equipment or systems that are credited in the full power internal events PRA. Systems are included based on an iterative process to include equipment that may be significant to the fire risk.	For this sensitivity case, equipment or cables assumed failed or credited by assumed routing in the base FPRA were assumed to always be available. The results demonstrate that FPRA results are sensitive to changes when the equipment is assumed available. However, a review of the sensitivity analysis results identified the reasons for the decrease in FPRA results were non-conservative given the contributing equipment is not expected to be available. The scope of credited equipment and cables and assumed cable routing is based on reviews of the applicable systems and the PRA model. Therefore, the scope credited equipment in the FPRA provides best estimate results.

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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 Fire PRA includes conservative adjustments to the HFEs to account for adverse impacts of fire events. The Fire PRA does not include credit for all operator actions, including fire response actions. The Fire PRA does not include credit for all instrument cues that may be available. A minimum joint HEP was applied for the HRA dependency analysis. Applying a minimum joint HEP may skew the results by artificially increasing the risk due to human failure events. The HEPs are propagated in the parametric uncertainty evaluation based on the uncertainty parameters from the Human Reliability Assessment Calculator (HRAC).	The use of 1E-6 is consistent with industry guidance for FPRA and is an accepted practice. A sensitivity case was performed for the base FPRA using a minimum joint HEP of 1E-5. The results demonstrate that using a higher FPRA minimum joint HEP has a slight impact on FPRA results. Further, as directed by NEI 00-04, the fire model human error basic events are increased to their 95^{th} percentile and also decreased to their 5^{th} percentile values as part of the required 50.69 PRA categorization sensitivity cases.