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RS-20-053

10 CFR 50.90
10 CFR 50.69

April 30, 2020

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

Clinton Power Station, Unit 1
Facility Operating License No. NPF-62
NRC Docket No. 50-461

Subject: Application to Adopt 10 CFR 50.69, "Risk-informed categorization and treatment of structures, systems, and components for nuclear power reactors"

In accordance with the provisions of 10 CFR 50.69 and 10 CFR 50.90, Exelon Generation Company, LLC (EGC) is requesting an amendment to Facility Operating License No. NPF 62 for Clinton Power Station (CPS), Unit 1.

The proposed amendment would modify the CPS licensing basis, by the addition of a License Condition, to allow for the implementation of the provisions of Title 10 of the Code of Federal Regulations (10 CFR), Part 50.69, "Risk-Informed categorization and treatment of structures, systems and components for nuclear power reactors." The provisions of 10 CFR 50.69 allow adjustment of the scope of equipment subject to special treatment controls (e.g., quality assurance, testing, inspection, condition monitoring, assessment, and evaluation). For equipment determined to be of low safety significance, alternative treatment requirements can be implemented in accordance with this regulation. For equipment determined to be of high safety significance, requirements will not be changed or will be enhanced. This allows improved focus on equipment that has safety significance resulting in improved plant safety.

The enclosure to this letter provides the basis for the proposed change to the CPS 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 EGC submittal of the LAR dated April 30, 2020, "Application to Revise Technical Specifications to Adopt Risk Informed Completion Times TSTF-505, Revision 2, 'Provide Risk-Informed Extended Completion Times – RITSTF Initiative 4b,'" (RS-20-052). EGC requests that the NRC conduct their review of the PRA technical adequacy details for this application in coordination with the review of the application currently in-process. This would reduce the number of EGC and NRC resources necessary to complete the review of the applications. This request should not be considered a linked requested licensing action, as the details of the PRA models in each LAR are complete which will allow the NRC staff to independently review and approve each LAR on their own merits without regard to the results from the review of the other.

EGC requests approval of the proposed license amendment by April 30, 2021, with the amendment being implemented within 60 days.

The proposed change has been reviewed by the Plant Operations Review Committee in accordance with the requirements of the EGC Quality Assurance Program.

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

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

There are no regulatory commitments contained in this letter. Should you have any questions concerning this letter, please contact Mr. Kenneth M. Nicely at (630) 657-2803.

I declare under penalty of perjury that the foregoing is true and correct. Executed on the 30th day of April 2020.

Respectfully,

Patrick R. Simpson

Patrick R. Simpson
Sr. Manager Licensing

Enclosure: Evaluation of the Proposed Change

cc: NRC Regional Administrator, Region III
NRC Senior Resident Inspector – Clinton Power Station
Illinois Emergency Management Agency – Division of Nuclear Safety

ENCLOSURE
Evaluation of the Proposed Change

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LIST OF ATTACHMENTS

1 SUMMARY DESCRIPTION

The proposed amendment modifies the licensing basis to allow for the implementation of the provisions of Title 10 of the Code of Federal Regulations (10 CFR), Part 50.69, "Risk-Informed Categorization and Treatment of Structures, Systems and Components for Nuclear Power Reactors." The provisions of 10 CFR 50.69 allow adjustment of the scope of equipment subject to special treatment controls (e.g., quality assurance, testing, inspection, condition monitoring, assessment, and evaluation). For equipment determined to be of low safety significance (LSS), alternative treatment requirements can be implemented in accordance with this regulation. For equipment determined to be of high safety significance (HSS), requirements will not be changed or will be enhanced. This allows improved focus on equipment that has safety significance resulting in improved plant safety.

2 DETAILED DESCRIPTION

2.1 CURRENT REGULATORY REQUIREMENTS

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

This deterministic approach establishes requirements for engineering margin and quality assurance in design, manufacture, and construction. In addition, it assumes that adverse conditions can exist (e.g., equipment failures and human errors) and establishes a specific set of design basis events (DBEs). The deterministic approach then requires that the facility include safety systems capable of preventing or mitigating the consequences of those DBEs to protect public health and safety. The Structures, Systems and Components (SSCs) necessary to defend against the DBEs are defined as "safety-related," and these SSCs are the subject of many regulatory requirements, herein referred to as "special treatments," designed to ensure that they are of high quality and high reliability, and have the capability to perform during postulated design basis conditions. Treatment includes, but is not limited to, quality assurance, testing, inspection, condition monitoring, assessment, evaluation, and resolution of deviations. The distinction between "treatment" and "special treatment" is the degree of NRC specification as to what must be implemented for particular SSCs or for particular conditions. Typically, the regulations establish the scope of SSCs that receive special treatment using one of three different terms: "safety-related," "important to safety," or "basic component." The terms "safety-related "and "basic component" are defined in the regulations, while "important to safety," used principally in the general design criteria (GDC) of Appendix A to 10 CFR Part 50, is not explicitly defined.

2.2 REASON FOR PROPOSED CHANGE

A probabilistic approach to regulation enhances and extends the traditional deterministic approach by allowing consideration of a broader set of potential challenges to safety, providing a logical means for prioritizing these challenges based on safety significance, and allowing consideration of a broader set of resources to defend against these challenges. In contrast to the deterministic approach, Probabilistic Risk Assessments (PRAs) address credible initiating events by assessing the event frequency. Mitigating system reliability is then assessed, including the potential for common cause failures. The probabilistic approach to regulation is an extension and enhancement of traditional regulation by considering risk in a comprehensive manner.

To take advantage of the safety enhancements available through the use of PRA, in 2004 the NRC published a new regulation, 10 CFR 50.69. The provisions of 10 CFR 50.69 allow adjustment of the scope of equipment subject to special treatment controls (e.g., quality assurance, testing, inspection, condition monitoring, assessment, and evaluation). For equipment determined to be of low safety significance, alternative treatment requirements can be implemented in accordance with the regulation. For equipment determined to be of high safety significance, requirements will not be changed or will be enhanced. This allows improved focus on equipment that has safety significance resulting in improved plant safety.

The rule contains requirements on how a licensee categorizes SSCs using a risk-informed process, adjusts treatment requirements consistent with the relative significance of the SSC, and manages the process over the lifetime of the plant. A risk-informed categorization process is employed to determine the safety significance of SSCs and place the SSCs into one of four risk-informed safety class (RISC) categories. The determination of safety significance is performed by an integrated decision-making process, as described by NEI 00-04, "10 CFR 50.69 SSC Categorization Guideline" (Reference [1]), which uses both risk insights and traditional engineering insights. The safety functions include the design basis functions, as well as functions credited for severe accidents (including external events). Special or alternative treatment for the SSCs is applied as necessary to maintain functionality and reliability and is a function of the SSC categorization results and associated bases. Finally, periodic assessment activities are conducted to make adjustments to the categorization and/or treatment processes as needed so that SSCs continue to meet all applicable requirements.

The rule does not allow for the elimination of SSC functional requirements or allow equipment that is required by the deterministic design basis to be removed from the facility. Instead, the rule enables licensees to focus their resources on SSCs that make a significant contribution to plant safety. For SSCs that are categorized as high safety significant, existing treatment requirements are maintained or enhanced. Conversely, for SSCs that do not significantly contribute to plant safety on an individual basis, the rule allows an alternative risk-informed approach to treatment that provides reasonable, though reduced, level of confidence that these SSCs will satisfy functional requirements.

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

2.3 DESCRIPTION OF THE PROPOSED CHANGE

EGC proposes the addition of the following condition to the operating license of CPS to document the NRC's approval of the use 10 CFR 50.69.

> Exelon Generation Company, LLC (EGC) is approved to implement 10 CFR 50.69 using the processes for categorization of Risk-Informed Safety Class (RISC)-1, RISC-2, RISC-3,

and RISC-4 Structures, Systems, and Components (SSCs) using: Probabilistic Risk Assessment (PRA) models to evaluate risk associated with internal events, including internal flooding, and internal fire; the shutdown safety assessment process to assess shutdown risk; the Arkansas Nuclear One, Unit 2 (ANO-2) passive categorization method to assess passive component risk for Class 2 and Class 3 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 EGC's submittal letter dated April 30, 2020, and all its subsequent associated supplements as specified in License Amendment No. [XXX] dated [DATE].

Prior NRC approval, under 10 CFR 50.90, is required for a change to the categorization process specified above (e.g., change from a seismic margins approach to a seismic probabilistic risk assessment approach).

3 TECHNICAL EVALUATION

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

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

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

The PRA models described within this license amendment request (LAR) are the same as those described within the EGC submittal of the LAR dated April 30, 2020, "Application to Revise Technical Specifications to Adopt Risk Informed Completion Times TSTF-505, Revision 2, 'Provide Risk-Informed Extended Completion Times – RITSTF Initiative 4b,'" (RS-20-052).

EGC requests that the NRC conduct their review of the PRA technical adequacy details for this application in coordination with the review of the application currently in-process. This would reduce the number of EGC and NRC resources necessary to complete the review of the applications. This request should not be considered a linked requested licensing action, as the details of the PRA models in each LAR are complete which will allow the NRC staff to independently review and approve each LAR on their own merits without regard to the results from the review of the other.

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

3.1.1 Overall Categorization Process

EGC will implement the risk categorization process in accordance with NEI 00-04, Revision 0, as endorsed by Regulatory Guide (RG) 1.201, "Guidelines for Categorizing Structures, Systems, and Components in Nuclear Power Plants According to their Safety Significance" [2]. NEI 00-04 Section 1.5 states "Due to the varying levels of uncertainty and degrees of conservatism in the spectrum of risk contributors, the risk significance of SSCs is assessed separately from each of five risk perspectives and used to identify SSCs that are potentially safety-significant." A separate evaluation is appropriate to avoid reliance on a combined result that may mask the results of individual risk contributors.

The process to categorize each system will be consistent with the guidance in NEI 00-04, "10 CFR 50.69 SSC Categorization Guideline," as endorsed by RG 1.201, with the exception of the evaluation of impact of the seismic hazard, which will use the EPRI 3002017583 (Reference [3]) approach for seismic Tier 1 sites, which includes CPS, to assess seismic hazard risk for 50.69. Inclusion of additional process steps discussed below to address seismic considerations will ensure that reasonable confidence in the evaluations required by 10 CFR 50.69(c)(1)(iv) is achieved. RG 1.201 states that "the implementation of all processes described in NEI 00-04 (i.e., Sections 2 through 12) is integral to providing reasonable confidence" and that "all aspects of NEI 00-04 must be followed to achieve reasonable confidence in the evaluations required by §50.69(c)(1)(iv)." However, neither RG 1.201 nor NEI 00-04 prescribe a particular sequence or order for each of the elements to be completed. Therefore, the order in which each of the elements of the categorization process (listed below) is completed is flexible and as long as they are all complete they may even be performed in parallel. Note that NEI 00-04 only requires Item 3 to be completed for components/functions categorized as Low Safety Significant (LSS) by all other elements. Similarly, NEI 00-04 only requires Item 4 to be completed for safety-related active components/functions categorized as LSS by all other elements.

1. PRA-based evaluations (e.g., the internal events, internal flooding, and fire PRAs)

2. non-PRA approaches (e.g., Fire Safe Shutdown Equipment List, Seismic Safe Shutdown Equipment List, other external events screening, and shutdown assessment)

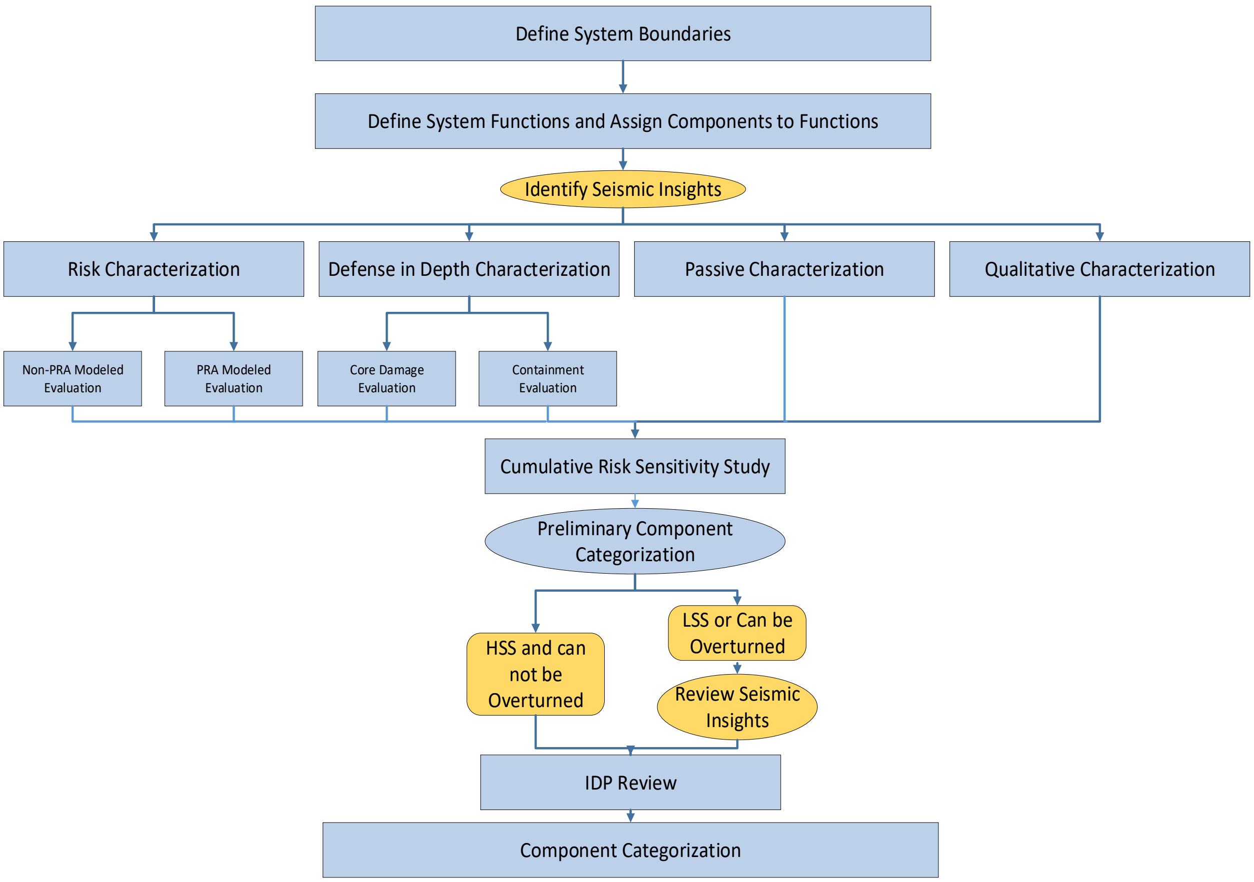
3. Seven qualitative criteria in Section 9.2 of NEI 00-04

4. the defense-in-depth assessment

5. the passive categorization methodology

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

Figure 3-1: Categorization Process Overview



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

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

Table 3-1: Categorization Evaluation Summary

Element	Categorization Step - NEI 00-04 Section	Evaluation Level	IDP Change HSS to LSS	Drives Associated Functions
Risk (PRA Modeled)	Internal Events Base Case – Section 5.1	Component	Not Allowed	Yes
	Fire, Seismic and Other External Events Base Case		Allowable	No
	PRA Sensitivity Studies		Allowable	No
	Integral PRA Assessment – Section 5.6		Not Allowed	Yes
Risk (Non-modeled)	Fire and Other External Hazards	Component	Not Allowed	No
	Seismic	Function/Component	Allowed [2]	No
	Shutdown – Section 5.5	Function/Component	Not Allowed	No
Defense-in-Depth	Core Damage – Section 6.1	Function/Component	Not Allowed	Yes
	Containment – Section 6.2	Component	Not Allowed	Yes
Qualitative Criteria	Considerations – Section 9.2	Function	Allowable[1]	N/A
Passive	Passive – Section 4	Segment/Component	Not Allowed	No

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

however the final assessments of the seven considerations are the direct responsibility of the IDP.

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

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

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

The mapping of components to system functions is used in some categorization process steps to facilitate preliminary categorization of components. Specifically, functions with mapped components that are determined to be HSS by the PRA-based assessment (i.e., Internal Events PRA or Integral PRA assessment) or defense-in-depth evaluation will be initially treated as HSS. However, NEI 00-04 Section 10.2 allows detailed categorization which can result in some components mapped to HSS functions being treated as LSS; and Section 4.0 discusses additional functions that may be identified (e.g., fill and drain) to group and consider potentially LSS components that may have been initially associated with a HSS function but which do not support the critical attributes of that HSS function. Note that certain steps of the categorization process are performed at a component level (e.g., Passive, Non-PRA-modeled hazards – see Table 3-1). Except for seismic, these components from the component level assessments will remain HSS (IDP cannot override) regardless of the significance of the functions to which they are mapped. Components having seismic functions may be HSS or LSS based on the IDP's consideration of the seismic insights applicable to the system being categorized. Therefore, if an HSS component is mapped to an LSS function, that component will remain HSS. If an LSS component is mapped to an HSS function, that component may be driven HSS based on Table 3-1 above or may remain LSS. For the seismic hazard, given that CPS is a seismic Tier 1 (low seismic hazard) plant as defined in Reference [3], seismic considerations are not required to drive an HSS determination at the component level, but the IDP will consider available seismic information pertinent to the components being categorized and can, at its discretion, determine that a component should be HSS based on that information.

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

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

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

- The decision criteria for the IDP for categorizing SSCs as safety significant or low safety-significant pursuant to § 50.69(f)(1) will be documented in EGC procedures.

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

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

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

- NEI 00-04 Section 7 requires assigning the safety significance of functions to be preliminary HSS if it is supported by an SSC determined to be HSS from the PRA-based assessment in Section 5 but does not require this for SSCs determined to be HSS from non-PRA-based, deterministic assessments in Section 5. This requirement is further clarified in the Vogtle SE (Reference [4]) which states "…if any SSC is identified as HSS from either the integrated PRA component safety significance assessment (Section 5 of NEI 00-04) or the defense-in-depth assessment (Section 6), the associated system function(s) would be identified as HSS."

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

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

- CPS 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 for Tier 1 plants and is discussed in Section 3.2.3.

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

- Internal Event Risks: Internal events including internal flooding PRA, as submitted to the NRC for TSTF-505 dated April 30, 2020, (RS-20-052) (Refer to Attachment 2).

- Fire Risks: Fire PRA model, as submitted to the NRC for TSTF-505 dated April 30, 2020, (RS-20-052) (Refer to 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 IPEEE screening process as approved by NRC SE dated December 6, 2000, (Reference (Clinton Power Station - Individual Examination of External Events, December 6, 2000 (TAC NO. M83607))). The other external hazards were determined to be insignificant contributors to plant risk.

- Low Power and Shutdown Risks: Qualitative defense-in-depth (DID) shutdown model for shutdown Configuration Risk Management (CRM) based on the framework for DID provided in NUMARC 91-06, "Guidance for Industry Actions to Assess Shutdown Management" (Reference [6]), which provides guidance for assessing and enhancing safety during shutdown operations.

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

1. Program procedures used in the categorization
2. System functions, identified and categorized with the associated bases
3. Mapping of components to support function(s)
4. PRA model results, including sensitivity studies
5. Hazards analyses, as applicable
6. Passive categorization results and bases

7. Categorization results including all associated bases and RISC classifications

8. Component critical attributes for HSS SSCs

9. Results of periodic reviews and SSC performance evaluations

10. IDP meeting minutes and qualification/training records for the IDP members

3.1.2 Passive Categorization Process

For the purposes of 10 CFR 50.69 categorization, passive components are those components that have a pressure retaining function. Passive components and the passive function of active components will be evaluated using the Arkansas Nuclear One (ANO) Risk-Informed Repair/Replacement Activities (RI-RRA) methodology contained in Reference [7] (ML090930246) consistent with the related Safety Evaluation (SE) issued by the Office of Nuclear Reactor Regulation.

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

The use of this method was previously approved to be used for a 10 CFR 50.69 application by NRC in the final Safety Evaluation for Vogtle dated December 17, 2014 (Reference [4]). The RI-RRA method as approved for use at Vogtle for 10 CFR 50.69 does not have any plant specific aspects and is generic. It relies on the conditional core damage and large early release probabilities associated with postulated ruptures. Safety significance is generally measured by the frequency and the consequence of the event. However, this RI-RRA process categorizes components solely based on consequence, which measures the safety significance of the passive component given that it ruptures. This approach is conservative compared to including the rupture frequency in the categorization as this approach will not allow the categorization of SSCs to be affected by any changes in frequency due to changes in treatment. The passive categorization process is intended to apply the same risk-informed process accepted by the NRC in the ANO2-R&R-004 for the passive categorization of Class 2, 3, and non-class components. This is the same passive SSC scope the NRC has conditionally endorsed in ASME Code Cases N-660 and N-662 as published in Regulatory Guide 1.147, Revision 15. Both code cases employ a similar risk-informed safety classification of SSCs in order to change the repair/ replacement requirements of the affected LSS components. All ASME Code Class 1 SSCs with a pressure retaining function, as well as supports, will be assigned high safety-significant, HSS, for passive categorization which will result in HSS for its risk-informed safety classification and cannot be changed by the IDP. Therefore, this methodology and scope for passive categorization is acceptable and appropriate for use at CPS 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 license amendment request (LAR) are the same as those described within the EGC submittal of the LAR dated April 30, 2020, "Application to Revise Technical Specifications to Adopt Risk Informed Completion Times TSTF-505, Revision 2, 'Provide Risk-Informed Extended Completion Times – RITSTF Initiative 4b,'" (RS-20-052).

3.2.1 Internal Events and Internal Flooding

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

3.2.2 Fire Hazards

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

3.2.3 Seismic Hazards

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

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

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

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

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

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

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

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

For example, the 10 CFR 50.69 categorization process as defined in NEI 00-04 includes an Integral Assessment that weighs the hazard specific relative importance of a component (e.g., internal events, internal fire, seismic) by the fraction of the total Core Damage Frequency (CDF) contributed by that hazard. The risk from an external hazard can be reduced from the default condition of HSS if the results of 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 CPS 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 [9]) 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 CPS, there is no need to perform more detailed evaluations to demonstrate the inherent seismic capacities documented in industry sources such as Reference [9]. Low seismic demand sites have lower likelihood of seismically-induced failures and lesser challenges to plant systems. This, therefore, provides the technical basis for allowing use of a graded approach for addressing seismic hazard at CPS.

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

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

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

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

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

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

Test cases described in Section 3 of Reference [3] showed that it would be unusual even for moderate hazard plants to exhibit any unique seismic insights, including due to correlated failures. The plant specific Reference [3] test case information Exelon is using from other licensees and being incorporated by Reference into this application is described in Case Study A (Reference [10]), Case Study C (References (Plant C Nuclear Plant, Units 1 and 2, License Amendment Request to Modify Approved 10 CFR 50.69 Categorization Process, June 22, 2017 (ML17173A875)), [12], and Case Study D (References [13], [14], [15]). Hence, while it is prudent to perform additional evaluations to identify conditions where correlated failures may occur for Tier 2 sites, for Tier 1 sites such as CPS, 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 CPS seismic hazard changes to medium risk (i.e., Tier 2) at some future time, CPS 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 CPS meets the Tier 1 site criteria.

In response to the NRC 50.54(f) letter associated with post-Fukushima recommendations (Reference [16]), CPS submitted a seismic hazard screening report (Reference [17]) to the NRC. The GMRS for CPS is below or approximately equal to the SSE between 1 Hz and 10 Hz and therefore meets the Tier 1 criterion in Reference [3].

The CPS SSE and GMRS curves from the seismic hazard and screening response in Reference [17] are shown in Figure A4-1. The NRC's staff assessment of the CPS seismic hazard and screening response is documented in Reference [18]. In section 3.4 of Reference [18] the NRC concluded that the methodology used by Exelon in determining the GMRS was acceptable and that the GMRS determined by Exelon adequately characterizes the reevaluated hazard for the CPS site.

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

1. NTTF Recommendation 2.1 seismic hazard screening (References [17], [18]), [19]).

2. NTTF Recommendation 2.3 seismic walkdowns (References [20], [21], [22], [23])

3. NTTF Recommendation 4.2 seismic mitigation strategy assessment (S-MSA) (References [24], [25])

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

4. NTTF Recommendation 2.1 seismic high frequency evaluation (References [26], [27])

As an enhancement to the EPRI study results as they pertain to CPS, the proposed CPS 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 CPS. For example, as part of the categorization team's preparation of the System Categorization Document (SCD) that is presented to the IDP, a section will be included in the SCD that summarizes the identified plant seismic insights pertinent to the system being categorized, and will also state the basis for applicability of the EPRI 3002017583 study and the bases for CPS 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 of the CPS 10 CFR 50.69 LAR) 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 CPS) are calculated per Section 5.6 of NEI 00-04, and components for which these measures exceed the specified criteria are preliminary HSS which cannot be changed to LSS.

For HSS SSCs uniquely identified by the CPS PRA models but having design-basis functions during seismic events or functions credited for mitigation and prevention of severe accidents caused by seismic events, these will be addressed using non-PRA based qualitative assessments in conjunction with any seismic insights provided by the PRA.

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

The categorization team will review available CPS 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 [3] to assess seismic hazard risk for 10 CFR 50.69 with the additional reviews discussed above will provide a process for categorization of RISC–1, RISC–2, RISC–3, and RISC–4 SSCs that satisfies the requirements of § 50.69(c).

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

3.2.4 Other External Hazards

All external hazards, except for seismic, were screened for applicability to CPS per a plant-specific evaluation in accordance with GL 88-20 (Reference [28]) and updated to use the criteria in ASME PRA Standard RA-Sa-2009. Attachment 4 provides a summary of the external hazards screening results. Attachment 5 provides a summary of the progressive screening approach for external hazards.

3.2.5 Low Power & Shutdown

Consistent with NEI 00-04, the CPS categorization process will use the shutdown safety management plan described in NUMARC 91-06 for evaluation of safety significance related to low power and shutdown conditions. The overall process for addressing shutdown risk is illustrated in Figure 5-7 of NEI 00-04.

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

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

3.2.6 PRA Maintenance and Updates

The EGC risk management process ensures that the applicable PRA models used in this application continues to reflect the as-built and as-operated plant for CPS. The process delineates the responsibilities and guidelines for updating the PRA models, and includes criteria for both regularly scheduled and interim PRA model updates. The process includes provisions for monitoring potential areas affecting the PRA models (e.g., due to changes in the plant, errors or limitations identified in the model, and industry operational experience) for assessing the risk impact of unincorporated changes, and for controlling the model and associated computer files. The process will assess the impact of these changes on the plant PRA model in a timely manner but no longer than once every two refueling outages. If there is a significant impact on the PRA model, the SSC categorization will be re-evaluated.

In addition, EGC will implement a process that addresses the requirements in NEI 00-04, Section 11, "Program Documentation and Change Control." The process will review the results of periodic and interim updates of the plant PRA that may affect the results of the categorization process. If the results are affected, adjustments will be made as necessary to the categorization or treatment processes to maintain the validity of the processes. In addition, any PRA model upgrades will be peer reviewed prior to implementing those changes in the PRA model used for categorization.

3.2.7 PRA Uncertainty Evaluations

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

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

In the overall risk sensitivity studies, EGC will utilize a factor of 3 to increase the unavailability or unreliability of LSS components consistent with that approved for Vogtle in Reference [4]. Consistent with the NEI 00-04 guidance, EGC will perform both an initial sensitivity study and a cumulative sensitivity study. The initial sensitivity study applies to the system that is being categorized. In the cumulative sensitivity study, the failure probabilities (unreliability and unavailability, as appropriate) of all LSS components modeled in all identified PRA models for all systems that have been categorized are increased by a factor of 3. This sensitivity study together with the periodic review process assures that the potential cumulative risk increase from the categorization is maintained acceptably low. The performance monitoring process monitors the component performance to ensure that potential increases in failure rates of categorized components are detected and addressed before reaching the rate assumed in the sensitivity study.

The detailed process of identifying, characterizing and qualitative screening of model uncertainties is found in Section 5.3 of NUREG-1855 and Section 3.1.1 of EPRI TR-1016737 (Reference [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.

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

Key CPS 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 CPS PRA model specific assumptions or sources of uncertainty.

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

The PRA models described in Section 3.2 has been assessed against RG 1.200, "An Approach for Determining the Technical Adequacy of Probabilistic Risk Assessment Results for Risk-Informed Activities," Revision 2 (Reference [30]), consistent with NRC RIS 2007-06.

The internal events PRA model was subject to a self-assessment and a full-scope peer review conducted in October 2009.

The Fire PRA model was subject to a self-assessment and a full-scope peer review conducted in April 2018.

Two finding closure reviews were conducted on the FPIE and FPRA PRA models in December 2018 and November 2019. 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 [31]) as accepted by NRC in the letter dated May 3, 2017 (ML17079A427) (Reference [32]). The results of this review have been documented and are available for NRC audit.

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

- Open items and disposition from the CPS RG 1.200 self-assessment.
- Open findings and disposition of the CPS peer reviews.

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

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

The CPS 10 CFR 50.69 categorization process will implement the guidance in NEI 00-04. The overall risk evaluation process described in the NEI guidance addresses both known degradation mechanisms and common cause interactions and meets the requirements of §50.69(b)(2)(iv). Sensitivity studies described in NEI 00-04 Section 8 will be used to confirm that the categorization process results in acceptably small increases to core damage frequency (CDF) and large early release frequency (LERF). The failure rates for equipment and initiating event frequencies used in the PRA include the quantifiable impacts from known degradation mechanisms, as well as other mechanisms (e.g., design errors, manufacturing deficiencies, and human errors). Subsequent performance monitoring and PRA updates required by the rule will continue to capture this data and provide timely insights into the need to account for any important new degradation mechanisms.

3.5 FEEDBACK AND ADJUSTMENT PROCESS

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

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

The performance monitoring process is described in EGC'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 assurance, and others have responsibilities for preparing and conducting various performance monitoring tasks that feed into this process. The intent of the performance monitoring reviews is to discover trends in component reliability; to help catch and reverse negative performance trends and take corrective action if necessary.

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

The checklist includes:

- A review of the impact on the System Categorization Document (SCD) for configuration changes that may impact a categorized system under 10 CFR 50.69.
- Steps to be performed if redundancy, diversity, or separation requirements are identified or affected. These steps include identifying any potential seismic interaction between

added or modified components and new or existing safety related or safe shutdown components or structures.

- Review of impact to seismic loading, safe shutdown earthquake (SSE) seismic requirements, as well as the method of combining seismic components.

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

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

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

Scheduled periodic reviews no longer than once every two refueling outages will evaluate new insights resulting from available risk information (i.e., PRA model or other analysis used in the categorization) changes, design changes, operational changes, and SSC performance. If it is determined that these changes have affected the risk information or other elements of the categorization process such that the categorization results are more than minimally affected, then the risk information and the categorization process will be updated. This scheduled review will include:

- A review of plant modifications since the last review that could impact the SSC categorization.

- A review of plant specific operating experience that could impact the SSC categorization.

- A review of the impact of the updated risk information on the categorization process results.

- A review of the importance measures used for screening in the categorization process.

- An update of the risk sensitivity study performed for the categorization.

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

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

4 REGULATORY EVALUATION

4.1 APPLICABLE REGULATORY REQUIREMENTS/CRITERIA

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

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

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

- Regulatory Guide 1.174, "An Approach for Using Probabilistic Risk Assessment in Risk-Informed Decisions on Plant-Specific Changes to the Licensing Basis," Revision 2, April 2015.

- Regulatory Guide 1.200, "An Approach for Determining the Technical Adequacy of Probabilistic Risk Assessment Results for Risk-Informed Activities," Revision 2, March 2009.

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

4.2 NO SIGNIFICANT HAZARDS CONSIDERATION ANALYSIS

EGC proposes to modify the licensing basis to allow for the voluntary implementation of the provisions of Title 10 of the Code of Federal Regulations (10 CFR), Part 50.69, "Risk-Informed Categorization and Treatment of Structures, Systems and Components for Nuclear Power Reactors." The provisions of 10 CFR 50.69 allow adjustment of the scope of equipment subject to special treatment controls (e.g., quality assurance, testing, inspection, condition monitoring, assessment, and evaluation). For equipment determined to be of low safety significance, alternative treatment requirements can be implemented in accordance with this regulation. For equipment determined to be of high safety significance, requirements will not be changed or will be enhanced. This allows improved focus on equipment that has safety significance resulting in improved plant safety.

EGC has evaluated whether or not a significant hazards consideration is involved with the proposed amendment(s) by focusing on the three standards set forth in 10 CFR 50.92, "Issuance of amendment," as discussed below:

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

   Response: No.

The proposed change will permit the use of a risk-informed categorization process to modify the scope of Structures, Systems and Components (SSCs) subject to NRC special treatment requirements and to implement alternative treatments per the regulations. The process used to evaluate SSCs for changes to NRC special treatment requirements and the use of alternative requirements ensures the ability of the SSCs to perform their design function. The potential change to special treatment requirements does not change the design and operation of the SSCs. As a result, the proposed change does not significantly affect any initiators to accidents previously evaluated or the ability to mitigate any accidents previously evaluated. The consequences of the accidents previously evaluated are not affected because the mitigation functions performed by the SSCs assumed in the safety analysis are not being modified. The SSCs required to safely shut down the reactor and maintain it in a safe shutdown condition following an accident will continue to perform their design functions.

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

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

   Response: No.

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

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

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

   Response: No.

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

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

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

4.3 CONCLUSIONS

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

5 ENVIRONMENTAL CONSIDERATION

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

6 REFERENCES

[1] NEI 00-04, "10 CFR 50.69 SSC Categorization Guideline," Revision 0, Nuclear Energy Institute," July 2005.

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

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

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

[5] Clinton Power Station - Individual Examination of External Events, December 6, 2000 (TAC NO. M83607).

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

[7] ANO SER Arkansas Nuclear One, Unit 2 - Approval of Request for Alternative AN02-R&R-004, Revision 1, "Request to Use Risk-Informed Safety Classification and Treatment for Repair/Replacement Activities in Class 2 and 3 Moderate and High Energy Systems," (TAC NO. MD5250) (ML090930246), April 22, 2009.

[8] Not used.

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

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

[11] Plant C Nuclear Plant, Units 1 and 2, License Amendment Request to Modify Approved 10 CFR 50.69 Categorization Process, June 22, 2017 (ML17173A875).

[12] Plant C Nuclear Plant, Units 1 and 2, "Issuance of Amendments Regarding Application of Seismic Probabilistic Risk Assessment Into the Previously Approved 10 CFR 50.69 Categorization Process (EPID L-2017-LLA-0248)," August 10, 2018 (ML18180A062).

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

[14] Plant D Nuclear Plant, Units 1 and 2, Seismic Probabilistic Risk Assessment Supplemental Information, April 10, 2018 (ML18100A966).

[15] Plant D Nuclear Plant, Units 1 and 2, Application to Adopt 10 CFR 50.69, "Risk-informed Categorization and Treatment of Structures, Systems, and Components for Nuclear Power Reactors," November 29, 2018 (ML18334A363).

[16] U.S. Nuclear Regulatory Commission, Request for Information Pursuant to Title 10 of the Code of Federal Regulations 50.54(F) Regarding Recommendations 2.1,2.3, And 9.3, of

the Near-Term Task Force Review of Insights from the Fukushima Dai-Ichi Accident, March 12, 2012 (ML12053A340).

[17] Seismic Hazard and Screening Report (CEUS Sites), "Response to NRC Request for Information Pursuant to 10 CFR 50.54(f) Regarding Recommendation 2.1 of the NTTF Review of the Fukushima Dai-ichi Accident," Clinton Power Station Unit 1, March 31, 2014 (ML14091A011).

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[41] CPS 1860.01C001, "Operations Department Cold Weather Preparations Checklist," Revision 8d, August 2, 2017.

[42] NRC Regulatory Guide 1.78, "Evaluating the Habitability of a Nuclear Power Plant Control Room During a Postulated Hazardous Chemical Release," Revision 1, December 2001 (ML013100014).

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

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[46] NRC NUREG-1855, "Guidance on the Treatment of Uncertainties Associated with PRAs in Risk-Informed Decision Making," Revision 1, March 2017 (ML17062A466).

[47] Clinton Power Station Probabilistic Risk Assessment Summary Notebook, CL-PRA-013, Revision 6, February 2020.

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[49] CL-MISC-033, "PRA Uncertainty Evaluation for RI Applications," Revision 0, April 2020.

[50] NRC Regulatory Guide 1.174, "An Approach for Using Probabilistic Risk Assessment in Risk-Informed Decisions on Plant-Specific Changes to the Licensing Basis," Revision 3, January 2018 (ML17317A256).

Attachment 1: List of Categorization Prerequisites

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

- Integrated Decision-Making Panel (IDP) member qualification requirements

- Qualitative assessment of system functions. System functions are qualitatively categorized as preliminary High Safety Significant (HSS) or Low Safety Significant (LSS) based on the seven criteria in Section 9 of NEI 00-04 (see Section 3.2). Any component supporting an HSS function is categorized as preliminary HSS. Components supporting, an LSS function are categorized as preliminary LSS.

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

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

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

- Risk sensitivity study. For PRA-modeled components, an overall risk sensitivity study is used to confirm that the population of preliminary LSS components results in acceptably small increases to core damage frequency (CDF) and large early release frequency (LERF) and meets the acceptance guidelines of Regulatory Guide 1.174.

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

- Documentation requirements per Section 3.1.1 of the enclosure.

Attachment 2: Description of PRA Models Used in Categorization

Unit	Model	Baseline CDF	Baseline LERF	Comments
Full Power Internal Events (FPIE) PRA Model				
1	CL117B Peer Reviewed Against RG 1.200 R2 in October 2009	3.3E-06	1.7E-07	2019 FPIE Application Specific Model
Fire (FPRA) Model				
1	CL117BF0 Peer Reviewed Against RG 1.200 R2 in April 2018	7.8E-05	5.3E-06	2019 Fire PRA Application Specific Model

Finding Number	Supporting Requirement(s)	Capability Category (CC)	Description	Disposition for 50.69
1-32	LE-E1 QU-C1 QU-C2 HR-H3 LE-E4 QU-A5 HR-G7	Not Met	CPS-PSA-004 Section 5.2 discusses the use of screening values used for HEPs in order to identify cutsets with dependent HEPs. However, only twelve of the over 100 basic events modeling post-initiator operator actions are listed in Table 5.2-1 as using screening values to identify dependency. Of these, six use a value of 1.0E-02 and one uses a value of 1.0E-03. The remaining five use a value of 0.1. It appears that all other HEPs are quantified with their nominal values. Use of such low probability values is likely to result in combinations of dependent HEPs being omitted by truncation values. Use of a sufficiently high value for HEPs is required by SR QU-C1 and not using a sufficiently high value would result in an inadequate assessment of dependent HEPs. (This F&O originated from SR HR-G7)	This issue has minimal impact on the 50.69 application since all risk-significant HRA dependencies are captured through the current methodology and results. A review of the CDF & LERF cutsets was performed to determine if any HRA dependent combinations exist without escalated dependent joint HEPs (i.e., they assume zero dependence and thus the HEPs are unaltered). Separately, a review of the combinations concluded that a majority of the unanalyzed dependent combinations are related to time-phased actions (i.e., early vs. late) where no additional dependency need be assigned between the actions because the time-phased calculations already reflect the impacts of those dependencies. A few legitimate dependent combinations were identified upon further review,

Finding Number	Supporting Requirement(s)	Capability Category (CC)	Description	Disposition for 50.69
				however, increasing the dependent joint HEPs for these groups does not substantially impact the overall risk results. Further justification for the chosen truncation level used in the HRA Dependency Analysis is required in a future model update. Therefore, this open item is primarily a documentation issue.
1-34	LE-E1 QU-C1 QU-C2 HR-H3 LE-E4 QU-A5 HR-G7	Not Met	Solving the PRA models with some HEPs at nominal can result in cutsets with multiple operator actions being truncated out or with the combined probability of all operator actions much below the 1E-6 or 5E-7 floor that the HRA notebook says is used. The peer review team quantified the PRA model with post-initiator HEPs set to 0.1 and identified a significant number of cutsets containing combinations of basic events representing operator action failure. These combinations were reviewed and a large number of combinations identified in this review were not included in the CPS HRA dependency evaluation. (This F&O originated from SR HR-G7)	See discussion for F&O 1-32.

Attachment 4: External Hazards Screening

External Hazard	Screening Result		
	Screened? (Y/N)	**Screening Criterion (Note a)**	**Comment**
Aircraft Impact	Y	PS2 PS4	In the IPEEE (Reference (Clinton Power Station Individual Plant Examination for External Events Final Report, September 1995)), a probabilistic bounding analysis was performed for aircraft. The median frequency of aircraft accidents which could lead to radiological consequences in excess of the exposure guidelines of 10 CFR Part 100 was calculated less than 1E-7/year **(PS2)**. Sections 2.2.2.5 and 3.5.1.6 of the USAR (Reference (Clinton Power Station Updated Safety Analysis Report (USAR), Revision 20, October 2018)) describe the airports and airways in the vicinity of the site. a. There is one federal Low Altitude Federal Airway with its centerline passing within 2 miles east of the station. Three additional Low Altitude Federal Airways within 6 miles were evaluated. The calculated frequency of aircraft is 0.54E-7/year (**PS4**). b. There are no commercial airports within 10 miles of the site. c. There are three private airstrips within 5 miles of the station. Based on this review, the Aircraft impact hazard can be considered to be negligible.

External Hazard	Screening Result		
	Screened? (Y/N)	Screening Criterion (Note a)	Comment
Avalanche	Y	C3	The mid-western location of CPS station precludes the possibility of an avalanche. Based on this review, the Avalanche hazard can be considered to be negligible.
Biological Event	Y	C5	Hazard is slow to develop and can be identified via monitoring and managed via standard maintenance process. Actions committed to and completed by CPS in response to Generic Letter 89-13 provide on-going control of biological hazards. These controls are described in EGC procedure ER-AA-340, "GL 89-13 Program Implementing Procedure" (Reference [35]). Based on this review, the Biological Event hazard can be considered to be negligible.
Coastal Erosion	Y	C3	The mid-western location of CPS station precludes the possibility of coastal erosion. Based on this review, the Coastal Erosion hazard can be considered to be negligible.
Drought	Y	C5	Drought is a slowly developing hazard allowing time for orderly plant reductions, including shutdowns. Based on this review, the Drought hazard can be considered to be negligible.

External Hazard	Screening Result		
	Screened? (Y/N)	Screening Criterion (Note a)	Comment
External Flooding	Y	C1	All non-LIP external flooding mechanisms were considered bounded by the plant's CDB. These mechanisms will not produce external flooding that will challenge any safety-related SSCs relied upon to safely shutdown the plant. External flooding from LIP will similarly not challenge any safety functions at CPS. The max WSE is calculated to be 1.2-inches above the building finished floor elevation at the Radwaste Building. In calculation IP-S-0282 and the Mitigating Strategies Flood Hazard Assessment (Reference [36]), it is shown that there are no SR SSCs in the Radwaste Building and water will not propagate or accumulate in any other buildings containing safety-related SSCs. Therefore, all external flooding mechanisms are screened and there are no SSCs credited for screening this hazard. Based on this review, the External Flooding hazard can be considered to be negligible.
Extreme Wind or Tornado	Y	C1	<u>Wind Hazard</u> Based on an evaluation in (Reference [37]), the plant design for wind pressure and the low frequency (<1E-7/yr) of design tornadoes, a demonstrably conservative estimate of CDF associated with high wind hazard (other than wind generated missiles) is much less than 1E-6/yr.

External Hazard	Screening Result		
	Screened? (Y/N)	Screening Criterion (Note a)	Comment
			Missile Hazard Based on an evaluation in (Reference [37]), the TORMIS-based calculation of the damage frequency for all SSCs unprotected against tornado missiles at the Clinton plant is 6.5E-7/yr. The TORMIS analysis determines the total arithmetic sum of the tornado induced missile damage frequency for the identified unprotected SSCs, but the analysis does not specifically calculate core damage frequency (CDF) or large early release frequency (LERF). However, given the conservatism in TORMIS analyses and the fact that multiple targets must be failed in order to cause core damage, the CDF associated with tornado missiles is estimated to be much less than 1E-6/yr. Based on this review, the Extreme Wind or Tornado hazard can be considered to be negligible.
Fog	Y	C4	Fog is discussed in the UFSAR Section 2.3 (Reference (Clinton Power Station Updated Safety Analysis Report (USAR), Revision 20, October 2018)). The principal effects of such events (such as freezing fog) would be to cause a loss of off-site power which is addressed in the weather-related Loss of Offsite Power initiating event in the internal events PRA model for CPS. Based on this review, the Fog hazard can be considered to be negligible.

External Hazard	Screening Result		
	Screened? (Y/N)	**Screening Criterion (Note a)**	**Comment**
Forest or Range Fire	Y	C3	Forest fires and grass fires were screened in the IPEEE (Reference (Clinton Power Station Individual Plant Examination for External Events Final Report, September 1995)). FSAR Section 2.2.3.1.4 also discusses this hazard and states that forest or brush fires cannot pose any danger because of the site landscaping. Based on this review, the Forest or Range Fire hazard can be considered to be negligible.
Frost	Y	C4	Frost is discussed in the UFSAR Section 2.5 (Reference (Clinton Power Station Updated Safety Analysis Report (USAR), Revision 20, October 2018)). The principal effects of such events would be to cause a loss of off-site power and are addressed in the weather-related Loss of Offsite Power initiating event in the internal events PRA model for CPS. Based on this review, the Frost hazard can be considered to be negligible.

External Hazard	Screening Result		
	Screened? (Y/N)	Screening Criterion (Note a)	Comment
Hail	Y	C4	Per the IPEEE (Reference (Clinton Power Station Individual Plant Examination for External Events Final Report, September 1995)), hailstorms need not be considered per the guidance contained in NUREG 1407 (Reference (NUREG 1407, June 1991)). The principal effects of such events would be to cause a loss of off-site power and are addressed in the weather-related Loss of Offsite Power initiating event in the internal events PRA model for CPS. Based on this review, the Hail hazard can be considered to be negligible.
High Summer Temperature	Y	C5	The principal effects of such events would result in elevated lake temperatures which are monitored during performance of control room shiftly checks. Should the ultimate heat sink temperature reach 93°F then operations' procedures require further increased monitoring and development of compensatory measures (Reference [39]). This phenomenon provides large amount of time for preparation (weather forecast) with time for implementation of appropriate mitigation actions. Based on this review, the High Summer Temperature hazard can be considered to be negligible.

External Hazard	Screening Result		
	Screened? (Y/N)	**Screening Criterion (Note a)**	**Comment**
High Tide, Lake Level, or River Stage	Y	C1 C5	High tide or river stage does not apply since there is only a cooling lake (Lake Clinton). Per USAR Section 2.4, the cooling lake (Lake Clinton) was formed by a dam with spillways to control high lake level **(C1)**. In addition, the event develops slowly, allowing adequate time to eliminate or mitigate the threat (**C5)**. See also "External Flooding." Based on this review, the High Tide, Lake Level, or River Stage hazard can be considered to be negligible.
Hurricane	Y	C3	The mid-western location of CPS precludes the possibility of a hurricane. Based on this review, the Hurricane hazard can be considered to be negligible.
Ice Cover	Y	C4	Per the IPEEE (Reference (Clinton Power Station Individual Plant Examination for External Events Final Report, September 1995)), ice storm need not be considered per the guidance contained in NUREG 1407 (Reference (NUREG 1407, June 1991)). The principal effects of such events would be to cause a loss of off-site power and are addressed in the weather-related Loss of Offsite Power initiating events in the internal events PRA model for CPS. Based on this review, the Ice Cover hazard can be considered to be negligible.

External Hazard	Screening Result		
	Screened? (Y/N)	**Screening Criterion (Note a)**	**Comment**
Industrial or Military Facility Accident	Y	C1 C3	Per FSAR Section 2.2.1 (Reference (Clinton Power Station Updated Safety Analysis Report (USAR), Revision 20, October 2018)), there are several military reserve unit armories located in the general area of the site which are listed in Table 2.2-1. The closest is the Bloomington armory located 23 miles NNW. The armories normally should contain no explosives. There are no military missile sites within 50 miles of the station **(C3)**. Per FSAR Section 2.2.3 (Reference (Clinton Power Station Updated Safety Analysis Report (USAR), Revision 20, October 2018)), no nearby industrial or other activities have been identified which could pose unusual hazards to the Clinton Power Station. All hazardous materials stored or shipped in the vicinity of CPS were evaluated in FSAR Subsection 2.2.3.1.3 for their toxic potential on control room habitability. Based on this evaluation, releases of hazardous materials in the vicinity of CPS are not considered as design basis accidents **(C1).** See also "Toxic Gas." Based on this review, the Industrial or Military Facility Accident hazard can be considered to be negligible.
Internal Flooding	N/A	N/A	The CPS Internal Events PRA includes evaluation of risk from internal flooding events.

External Hazard	Screening Result		
	Screened? (Y/N)	**Screening Criterion (Note a)**	**Comment**
Internal Fire	N/A	N/A	The CPS Internal Fire PRA includes evaluation of risk from internal fire events
Landslide	Y	C3	The mid-western location of CPS precludes the possibility of a landslide. Based on this review, the Landslide hazard can be considered to be negligible.
Lightning	Y	C1 C4	Per the IPEEE (Reference (Clinton Power Station Individual Plant Examination for External Events Final Report, September 1995)), the design of CPS includes features to protect against lightning **(C1)**. Lightning strikes are not uncommon in nuclear plant experience. They can result in losses of off-site power or surges in instrumentation output if grounding is not fully effective. The latter events often lead to reactor trips. Both events are incorporated into the CPS internal events model through the incorporation of generic and plant data **(C4)**. Based on this review, the Lightning hazard can be considered to be negligible.

External Hazard	**Screening Result**		
	Screened? (Y/N)	**Screening Criterion (Note a)**	**Comment**
Low Lake Level or River Stage	Y	C1 C5	Per USAR Section 2.4, the cooling lake (Lake Clinton) was formed by a dam with outlet works provided to control low lake level **(C1)**. In addition, the effect of low lake level would take place slowly allowing time for orderly plant reductions, including shutdowns **(C5)**. Based on this review, the Low Lake Level or River Stage hazard can be considered to be negligible.
Low Winter Temperature	Y	C5	Per the IPEEE (Reference (Clinton Power Station Individual Plant Examination for External Events Final Report, September 1995)), low winter temperature need not be considered per the guidance contained in NUREG 1407 (Reference (NUREG 1407, June 1991). However, there are existing severe weather procedures (Reference [40]) and cold weather checklists (Reference (CPS 1860.01C001, Revision 8d, August 2, 2017)) that are performed in advance of the onset of cold weather to allow adequate time to eliminate or mitigate the threat. Based on this review, the Low Winter Temperature hazard can be considered to be negligible.

External Hazard	Screening Result		
	Screened? (Y/N)	Screening Criterion (Note a)	Comment
Meteorite or Satellite Impact	Y	PS4	Per the IPEEE (Reference (Clinton Power Station Individual Plant Examination for External Events Final Report, September 1995)), meteorite or satellite need not be considered per the guidance contained in NUREG 1407 (Reference (NUREG 1407, June 1991)). However, the frequency of a meteor or satellite strike is judged to be so low as make the risk from such events insignificant. Based on this review, the Meteorite or Satellite hazard can be considered to be negligible.
Pipeline Accident	Y	C1	Per USAR Table 2.2-4 (Reference (Clinton Power Station Updated Safety Analysis Report (USAR), Revision 20, October 2018)) there is one pipeline within 5 miles of the site (at about 4650 feet) that transports refined petroleum products. Per USAR Section 2.2.3.1.1 the distance of 4600 feet has been established as a limit beyond which a possible pipeline rupture followed by an explosion under conservative weather conditions does not govern the design of the plant. Since the pipelines that existed prior to construction of the plant have been relocated (USAR Section 2.2.2.3) and the closest pipeline passes about 4650 feet from the site, explosions do not pose any hazard to the plant. Based on this review, the Pipeline Accident hazard can be considered to be negligible.

External Hazard	Screening Result		
	Screened? (Y/N)	Screening Criterion (Note a)	Comment
Release of Chemicals in Onsite Storage	Y	C1	Compliance with Regulatory Guide 1.78 (Reference [42]) for hazardous chemicals stored onsite is described in USAR Section 6.4 (Reference (Clinton Power Station Updated Safety Analysis Report (USAR), Revision 20, October 2018)). Gaseous chlorine is no longer allowed on site by plant procedure and there are no other significant depots of chlorine within a five mile radius of the site. Of the other potentially hazardous chemicals stored on site, listed in USAR Table 2.2-6, only sulfuric acid, carbon dioxide, and nitrogen are included in Regulatory Guide 1.78. The following are features protecting against potential problems upon a release of sulfuric acid: a. Sulfuric acid has a low vapor pressure (< 1 Torr), b. The relative location of the sulfuric acid storage facility with respect to the control room minimum outside air intakes, and c. The acid storage tank is vented to the outside. Fumes from spillage within the acid storage area are diluted by the exhaust air from the sulfuric acid storage area with the radwaste building and balance of the plant exhaust air streams. Analysis has shown that a postulated rupture in the carbon dioxide storage system does not result in an unacceptable concentration of CO2 within the control room. Since the amount of nitrogen stored onsite is not a significant fraction of the control

External Hazard	Screening Result		
	Screened? (Y/N)	**Screening Criterion (Note a)**	**Comment**
			room volume, per Regulatory Guide 1.78, it does not need to be considered. Based on this review, the Release of Chemicals in Onsite Storage hazard can be considered to be negligible.
River Diversion	Y	C3	Per UFSAR 2.4.1.2 (Reference (Clinton Power Station Updated Safety Analysis Report (USAR), Revision 20, October 2018)), the Salt Creek River is the principal tributary of the Sangamon River, which drains into the Illinois River. Per UFSAR 2.4.9 , there is no historical evidence of channel diversion of Salt Creek and North Fork of Salt Creek upstream of the dam site. Based on this review, the River Diversion hazard can be considered to be negligible.
Sand or Dust Storm	Y	C1	Per the IPEEE (Reference (Clinton Power Station Individual Plant Examination for External Events Final Report, September 1995)), sand or dust storm need not be considered per the guidance contained in NUREG 1407 (Reference (NUREG 1407, June 1991)). Based on this review, the Sand or Dust Storm hazard can be considered to be negligible.

External Hazard	Screening Result		
	Screened? (Y/N)	Screening Criterion (Note a)	Comment
Seiche	Y	C1	Per USAR Section 2.4.5, there is no large body of water near the site where significant seiche formations can occur. The size of the cooling lake is not large enough to develop a seiche flooding condition which is more critical than the probable maximum flood (PMF) condition. See also "External Flooding". Based on this review, the Seiche hazard can be considered to be negligible.
Seismic Activity	N	N/A	See Section 3.2.3 and Figure A4-1 in this Attachment.
Snow	Y	C1	Per the IPEEE (Reference (Clinton Power Station Individual Plant Examination for External Events Final Report, September 1995)), snow storm need not be considered per the guidance contained in NUREG 1407 (Reference (NUREG 1407, June 1991)). Based on this review, the Snow hazard can be considered to be negligible.

External Hazard	Screening Result		
	Screened? (Y/N)	**Screening Criterion (Note a)**	**Comment**
Soil Shrink-Swell Consolidation	Y	C1	USAR Chapter 2 (Reference (Clinton Power Station Updated Safety Analysis Report (USAR), Revision 20, October 2018)) discusses site characteristics and stability of soils. Extensive geotechnical investigations carried out prior to and during construction (including geologic mapping of the excavations) showed nothing that would preclude safe construction or operation of a nuclear-fueled power station. There are no known faults or folds of design significance at or anywhere near the site. Based on this review, the Soil Shrink-Swell Consolidation hazard can be considered to be negligible.
Storm Surge	Y	C3	The mid-western location of CPS precludes the possibility of a sea level driven storm surge. Based on this review, the Storm Surge hazard can be considered to be negligible.
Toxic Gas	Y	C1 C3	UFSAR Section 2.2.3.1.3 (Reference (Clinton Power Station Updated Safety Analysis Report (USAR), Revision 20, October 2018)) discusses toxic gas. Van Horn - DeWitt is the only facility within five miles of the site which manufactures, uses, or stores toxic chemicals. Van Horn - DeWitt is a distributor of agricultural products and chemicals (such as pesticides, herbicides, and fertilizers) and their facility in DeWitt is located

External Hazard	Screening Result		
	Screened? (Y/N)	**Screening Criterion (Note a)**	**Comment**
			approximately 2.5 miles from Clinton Power Station. CPS reviewed a list of chemicals distributed by Van Horn - DeWitt, and determined that with the exception of anhydrous ammonia, none of the chemicals require evaluation for their potential effect on control room habitability (due to an accidental spill or release) in accordance with Regulatory Guide 1.78. Calculations (Reference [42])) show the postulated accidents of the anhydrous ammonia nurse tanks and tanker trucks used by farmers and suppliers do not adversely affect the control room habitability **(C1)**. Reference (2013 Clinton Power Station Hazardous Chemical Survey, VC-94, Revision 0) concluded that all the identified toxic chemicals (transported via roadways) do not need further evaluation. In addition, Per FSAR Section 6.4.4.2, gaseous chlorine is no longer allowed on site by plant procedure and there are no other significant depots of chlorine within a five mile radius of the site. Therefore, no automatic initiation of the control room ventilation chlorine mode and no chlorine detectors are required **(C3)**. See also Release of Chemicals in onsite storage, industrial or military facility accident and transportation Accidents. Based on this review, the Toxic Gas hazard can be considered to be negligible.

External Hazard	Screening Result		
	Screened? (Y/N)	**Screening Criterion (Note a)**	**Comment**
Transportation Accident	Y	C1 PS2	Transportation accidents was evaluated in the IPEEE (Reference (Clinton Power Station Individual Plant Examination for External Events Final Report, September 1995)) and in USAR Section 2.2 (Reference (Clinton Power Station Updated Safety Analysis Report (USAR), Revision 20, October 2018)). In the IPEEE, the evaluation was conducted against the NRC Standard Review Plan which concluded that the risk was acceptably low **(PS2)**. Per FSAR 2.2.3 (Reference (Clinton Power Station Updated Safety Analysis Report (USAR), Revision 20, October 2018)), no nearby industrial or other activities have been identified which could pose unusual hazards to the Clinton Power Station. The nearest highway is State Highway 54 which passes about three-quarters of a mile from the reactor containment building. U.S. Highway 51, is approximately 6 miles from the site. The nearest railroad is the Gilman Line of the Canadian National/Illinois Central Railroad which runs parallel to Highway 54 and traverses north of the site approximately .75 miles. Effects of accidents on these transportation routes have been evaluated and it is concluded that they need not be considered as design basis events. The station is not located near a navigable waterway **(C1)**. Based on this review, the Transportation Accident hazard can be considered to be negligible.

External Hazard	Screening Result		
	Screened? (Y/N)	Screening Criterion (Note a)	Comment
Tsunami	Y	C3	The mid-western location of CPS precludes the possibility of a tsunami. Based on this review, the Tsunami hazard can be considered to be negligible.
Turbine-Generated Missiles	Y	PS4	Turbine generated missiles are discussed in UFSAR Section 3.5.1.3 (Reference (Clinton Power Station Updated Safety Analysis Report (USAR), Revision 20, October 2018)). With the replacement of the Low Pressure (LP) rotors, all the turbine rotors are of the monoblock design. The monoblock rotors have very low stress level. Missile generation due to turbine failure is generally postulated to be caused by turbine overspeed. General Electric has established that the speed capability of these rotors is considerably higher than the maximum attainable speed of these turbine generator units. Consequently, the probability of missiles being generated is statistically insignificant. Based on this review, the Turbine-Generated Missiles hazard can be considered to be negligible.
Volcanic Activity	Y	C3	Not applicable to the site because of location (no active or dormant volcanoes located near plant site). Based on this review, the Volcanic Activity hazard can be considered to be negligible.

<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>Waves</td><td>Y</td><td>C1</td><td>Waves associated with adjacent large bodies of water are not applicable to the site. Waves associated with external flooding are covered under that hazard.
Based on this review, the Waves hazard can be considered to be negligible.</td></tr>
<tr><td colspan="4">Note a – See Attachment 5 for descriptions of the screening criteria.</td></tr>
</table>

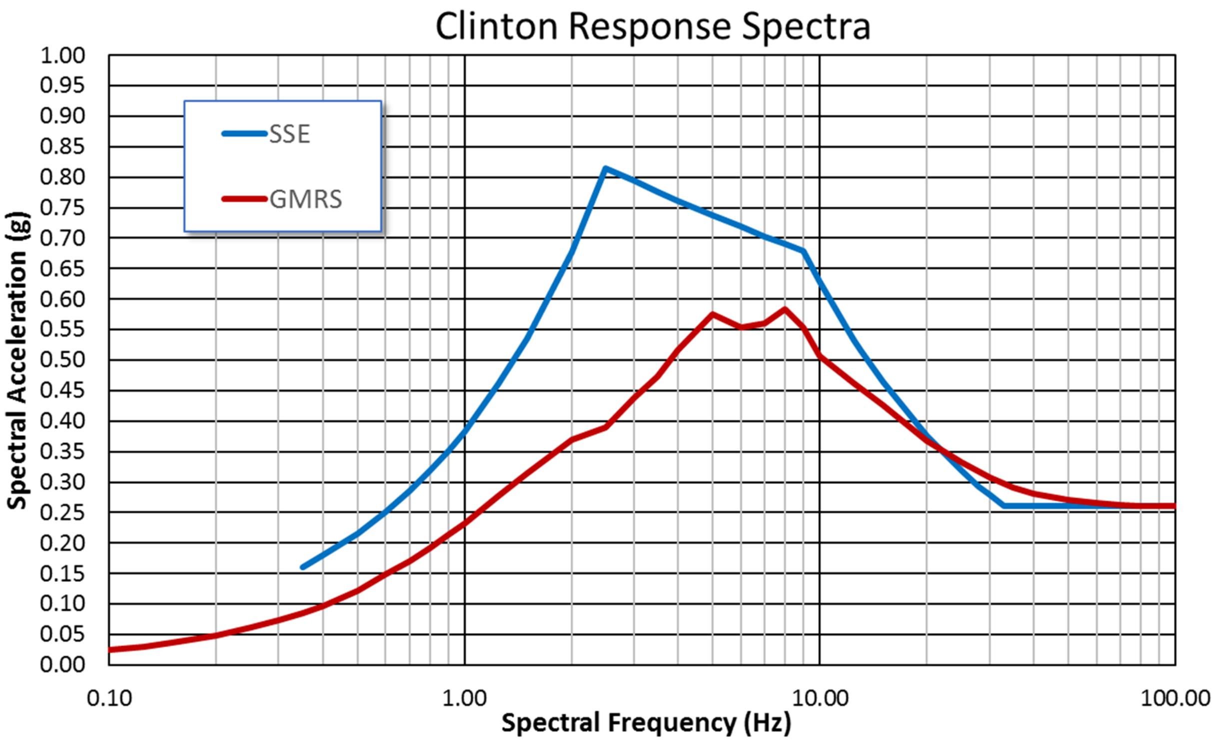

Figure A4-1: GMRS and SSE Response Spectra for CPS
(From Reference [17], Table 2.4-1 (GMRS) and Table 3.1-1 (SSE)

Attachment 5: Progressive Screening Approach for Addressing External Hazards

Event Analysis	Criterion	Source	Comments
Initial Preliminary Screening	C1. Event damage potential is < events for which plant is designed.	NUREG/CR-2300 and ASME/ANS Standard RA-Sa-2009	
	C2. Event has lower mean frequency and no worse consequences than other events analyzed.	NUREG/CR-2300 and ASME/ANS Standard RA-Sa-2009	
	C3. Event cannot occur close enough to the plant to affect it.	NUREG/CR-2300 and ASME/ANS Standard RA-Sa-2009	
	C4. Event is included in the definition of another event.	NUREG/CR-2300 and ASME/ANS Standard RA-Sa-2009	Not used to screen. Used only to include within another event.
	C5. Event develops slowly, allowing adequate time to eliminate or mitigate the threat.	ASME/ANS Standard RA-Sa-2009	
Progressive Screening	PS1. Design basis hazard cannot cause a core damage accident.	ASME/ANS Standard RA-Sa-2009	
	PS2. Design basis for the event meets the criteria in the NRC 1975 Standard Review Plan (SRP).	NUREG-1407 and ASME/ANS Standard RA-Sa-2009	
	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 CPS Internal Events and Fire PRA (FPRA) models and documentation were reviewed for generic (using the applicable lists of EPRI-identified generic sources of uncertainty per EPRI 1016737 (Reference [29]) and EPRI 1026511 (Reference [44]) and plant-specific modeling assumptions and related sources of uncertainty. 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 PRA Standard (Reference [45]) requirements for identification and characterization of uncertainties and assumptions. This evaluation identifies those sources of uncertainty that are important to the PRA results and may be important to PRA applications. The process meets the intent of steps C-1 and E-1 of NUREG-1855 (Reference [46]).

These evaluations are documented for internal events and internal flooding in the Summary Notebook (CL-PRA-013) (Reference [47]) and for internal fire in the Fire PRA Uncertainty and Sensitivity Notebook (CL-PRA-021.12) (Reference [48]). The results of the base PRA evaluations were reviewed to determine which potential uncertainties could impact the 50.69 program. This evaluation meets the intent of the screening portion of steps C-2 and E-2 of NUREG-1855 (Reference [46]).

Additionally, an evaluation of Level 2 Internal Events PRA model uncertainty was performed, based on the guidance in NUREG-1855 and Electric Power Research Institute (EPRI) report 1026511. The potential sources of model uncertainty in the CPS PRA model were evaluated for the 32 Level 2 PRA topics outlined in EPRI 1026511 which is documented in Section 5 of Reference [49].

FPIE - Disposition of Key Assumptions/Sources of Uncertainty

In order to identify key sources of uncertainty, the Internal Events PRA model uncertainties were evaluated using the guidance in NUREG-1855 (Reference [46]) and EPRI 1016737 (Reference [29]). As described in NUREG-1855, sources of uncertainty include "parametric" uncertainties, "modeling" uncertainties, and "completeness" (or scope and level of detail) uncertainties.

Parametric uncertainty was addressed as part of the Clinton PRA model quantification. The parametric uncertainty evaluation for the Internal Events PRA model is documented in Appendix B of the Summary Notebook (Reference [47]).

Modeling uncertainties are considered in both the base PRA and in specific risk-informed applications. Assumptions are made during the PRA development to address modeling uncertainties because there is not a single definitive approach. Plant-specific assumptions and modeling uncertainties for each of the CPS Internal Events PRA technical elements are noted in Appendix B of the Summary Notebook (Reference [47]). The Internal Events PRA model uncertainties evaluation considers the modeling uncertainties for the base PRA by identifying assumptions, determining if those assumptions are related to a source of modeling uncertainty and characterizing that uncertainty, as necessary. Electric Power Research Institute (EPRI) compiled a listing of generic sources of modeling uncertainty to be considered for each Internal Events PRA technical element (Reference [29]), and the evaluation performed for CPS considered each of the generic sources of modeling uncertainty as well as the plant-specific sources.

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

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 (Reference [50]) 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 as described above are evaluated relative to the 50.69 application in Attachment 6 to determine if additional sensitivity evaluations are needed.

> *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 error and common cause basic events are increased to their 95th percentile and also decreased to their 5th percentile values. These results are capable of driving a component and respective functions HSS and therefore the uncertainty of the PRA modeled HEPs and CCFs are accounted for in the 50.69 application.

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

IE / IF PRA Sources of Assumption/ Uncertainty	IE / IF PRA 50.69 Impact	IE / IF PRA Model Sensitivity and Disposition (50.69)
Impact of containment venting on core cooling system NPSH Many BWR core cooling systems utilize the suppression pool as a water source. Venting of containment as a decay heat removal mechanism or containment failure can substantially reduce NPSH, even lead to flashing of the pool. This rapid drop in containment pressure may lead to local steaming that causes steam binding in pumps taking suction on the suppression pool. The treatment of such scenarios varies across BWR PRAs.	ECCS is not credited for success after uncontrolled containment venting or induced containment failure (all failure mechanisms treated probabilistically). Upon initiation of uncontrolled containment venting or large containment failure, it is assumed that NPSH is lost for all systems taking suction from the suppression pool. Deterministic thermal hydraulic calculations are performed to support the controlled venting success criteria.	No credit for these systems after uncontrolled containment venting or large containment failure represents a slight conservative bias based on thermal hydraulic analyses. The PRA employs a reasonable and accepted approach. Therefore, this does not represent a key source of uncertainty for the CPS 50.69 application.

IE / IF PRA Sources of Assumption/ Uncertainty	IE / IF PRA 50.69 Impact	IE / IF PRA Model Sensitivity and Disposition (50.69)
Core cooling success following containment failure or venting through non-hard pipe vent paths Loss of containment heat removal leading to long-term containment over-pressurization and failure can be a significant contributor in some PRAs. Consideration of the containment failure mode might result in additional mechanical failures of credited systems. Containment venting through "soft" ducts or containment failure can result in loss of core cooling due to environmental impacts on equipment in the reactor building, loss of NPSH on ECCS pumps, steam binding of ECCS pumps, or damage to injection piping or valves. There is no definitive reference on the proper treatment of these issues.	1) LPCI and Core Spray are not credited for success after containment failure unless adverse conditions do not exist in the Auxiliary Building. Low pressure injection sources internal to containment (LPCI and LPCS from the suppression pool) are probabilistically evaluated to be available before containment failure. Low pressure injection sources internal to containment are also assumed to fail after containment failure due to the items listed in the discussion of the issue. 2) FW / Condensate / HPCS / SX and CRD are credited for success after containment failure, but an additional basic event is included that represents the likelihood that the containment failure size and location disrupt the capability of FW / Condensate / HPCS / SX and CRD to inject. Following containment failure, injection from CRD, FW/Condensate, and HPCS 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 III containment design and large-scale ultimate failure testing of steel containments.	Minimal credit for these systems after containment failure may represent a slight conservative bias. The PRA employs a reasonable and accepted approach. Therefore, this does not represent a key source of uncertainty for the CPS 50.69 application.

IE / IF PRA Sources of Assumption/ Uncertainty	IE / IF PRA 50.69 Impact	IE / IF PRA Model Sensitivity and Disposition (50.69)
Core melt arrest in-vessel Typically, the treatment of core melt arrest in-vessel has been limited. However, recent NRC work has indicated that there may be more potential than previously credited. A possible example is credit for CRD in BWRs as fully capable of arresting core melt progression in-vessel per MELCOR calculations.	1) The most likely scenarios for terminating in-vessel core melt progression are for high pressure core damage sequences with subsequent successful RPV depressurization. Therefore, high pressure core damage scenarios with subsequent RPV depressurization following core damage determine the likelihood of core melt arrest in-vessel. 2) Injection from these high capacity low pressure systems will preclude vessel failure if they are available following RPV depressurization but before the time at which vessel breach cannot be precluded given core damage occurs at high RPV pressure.	Core melt arrest prior to vessel failure may be credited to some degree with LP injection recovered after core damage, but prior to vessel failure. However, credit for the in-vessel arrest is limited to only a short amount of in-vessel core melt progression. The credit for in-vessel recovery has a slight conservative bias. The PRA employs a reasonable and accepted approach. Therefore, this does not represent a key source of uncertainty for the CPS 50.69 application.
Vessel failure mode The progression of core melt to the point of vessel failure remains uncertain. Some codes (MELCOR) predict that even vessels with lower head penetrations will remain intact until the water has evaporated from above the relocated core debris. Other codes (MAAP), predict that lower head penetrations might fail early. The failure mode of the vessel and associated timing can impact LERF determination, and may influence DCH characteristics (especially for some BWRs and PWR ice condenser plants).	Ex-vessel core melt progression overwhelming vapor suppression is explicitly considered in model for low pressure RPV failure sequences and high pressure RPV failure sequences. Ex-vessel core melt progression overwhelming vapor suppression is noted as extremely unlikely for low pressure RPV failures modes and very unlikely for high pressure failure modes based on reference to generic studies and identification of plant-specific features.	The values utilized for Ex-vessel core melt progression provide a reasonable best-estimate that will have minimal impact on the 50.69 calculations. The PRA employs a reasonable and accepted approach. Therefore, this does not represent a key source of uncertainty for the CPS 50.69 application.

IE / IF PRA Sources of Assumption/ Uncertainty	IE / IF PRA 50.69 Impact	IE / IF PRA Model Sensitivity and Disposition (50.69)
Treatment of Hydrogen combustion in BWR Mark III and PWR ice condenser plants The amount of hydrogen burned, the rate at which it is generated and burned, the pressure reduction mitigation credited by the suppression pool, ice condenser, structures, etc., can have a significant impact on the accident sequence progression development.	This failure mode is a moderate contributor to the LERF risk profile. The Mark III containment is not inerted. Hydrogen igniters are provided for controlled burn of hydrogen buildup in containment. Severe accident progression is modeled to lead to hydrogen combustion which fails containment with operation of the hydrogen igniters.	Slightly conservative assessment of hydrogen combustion. The values utilized for hydrogen combustion provide a reasonable best-estimate that will have minimal impact on the 50.69 calculations. The PRA employs a reasonable and accepted approach. Therefore, this does not represent a key source of uncertainty for the CPS 50.69 application.
Digital feedwater control failure There are model uncertainties associated with modeling digital systems, such as those related to determining the failure modes of these systems and components.	The reliability analysis for causing plant trips performed by FW vendor studies is assumed to be equally applicable to the reliability of the system post plant trips that are caused by other means that do not directly affect the feedwater availability. The reliability values from the vendor study demonstrating that the system performance would result in less than 0.1 transients per year are used for the key components of the system.	Digital feedwater control failure events are treated probabilistically. The PRA employs a reasonable and accepted approach. Therefore, this does not represent a key source of uncertainty for the CPS 50.69 application.
Credit for CRD following containment challenges Adverse Auxiliary Building (AB) conditions could arise as a result of containment failure. However, the CRD pumps are in the Turbine Building. Therefore, only failure modes that would directly impact the CRD injection lines are modeled to fail CRD following containment failure.	Credit for CRD pumps in many of the Class II type challenges. CRD pumps survive a small containment failure.	The PRA employs a reasonable and accepted approach. Therefore, this does not represent a key source of uncertainty for the CPS 50.69 application.

IE / IF PRA Sources of Assumption/ Uncertainty	IE / IF PRA 50.69 Impact	IE / IF PRA Model Sensitivity and Disposition (50.69)
FLEX Equipment Reliability There are no industry-approved data sources for FLEX equipment reliability.	FLEX is credited for SBO / ELAP sequences. The equipment failure rates from similar non-FLEX systems are doubled as surrogated for the FLEX equipment (until industry-approved FLEX data is developed) (e.g., failure rates for emergency DGs are doubled for the FLEX DGs). Also, a bounding value of 0.1 is included in the logic to account for potential random FLEX failure given successful implementation. Given the bounding value used for random failure of FLEX given successful implementation, any changes to the FLEX equipment reliability data are not expected to significantly impact the results.	The PRA employs a reasonable approach (industry guidance is still in development). However, FLEX may represent a key source of uncertainty for the CPS 50.69 application and will be further evaluated as part of implementation of the 50.69 program.
Water Hammer Water hammer is a potential failure mode for ECCS that may result in a large flood in the Auxiliary Building (AB) basement.	Water hammer consequences include flow blockage, system leakage, or system rupture. ECCS draindown scenarios are included in the PRA model. Subsequent starting / restarting of systems modeled as susceptible to water hammer (with or without starting water leg pumps) can cause a water hammer and flooding of the AB basement. The water hammer consequences are treated probabilistically based on an evaluation performed for LaSalle County Generating Station (LAS). The evaluation performed for LAS was judged applicable to CPS. (Note that similar approaches are generally employed in other EGC PRAs).	The water hammer evaluation is treated probabilistically. The PRA employs a reasonable and accepted approach. Therefore, this does not represent a key source of uncertainty for the CPS 50.69 application.

IE / IF PRA Sources of Assumption/ Uncertainty	IE / IF PRA 50.69 Impact	IE / IF PRA Model Sensitivity and Disposition (50.69)
Containment integrity following vessel rupture event There is model uncertainty regarding the subsequent treatment that increases the likelihood of LERF for this extremely rare event.	The scenarios that result in early containment failure are classified as Accident Class 3D scenarios with a high potential for LERF. A portion of the vessel rupture sequences are assumed to result in concurrent containment failure coincident with the vessel rupture.	Containment integrity following vessel rupture is treated probabilistically. The PRA employs a reasonable and accepted approach. Therefore, this does not represent a key source of uncertainty for the CPS 50.69 application.

Fire PRA - Disposition of Key Assumptions/Sources of Uncertainty

The Fire PRA (FPRA) model includes various sources of uncertainty that exist because there is both inherent randomness in elements that comprise the FPRA, and because the state of knowledge in these elements continues to evolve. The development of the FPRA was guided by NUREG/CR-6850 [9].

In order to identify key sources of uncertainty, the FPRA model uncertainty report was developed, based on the guidance in NUREG-1855 [1] and EPRI 1026511 [3]. As described in NUREG-1855, sources of uncertainty include "parametric" uncertainties, "modeling" uncertainties, and "completeness" (or scope and level of detail) uncertainties.

Parametric uncertainty was addressed as part of the Clinton FPRA model quantification. The parametric uncertainty evaluation for the FPRA model is documented in Appendix A of the Uncertainty and Sensitivity Analysis Notebook [6].

Modeling uncertainties are considered in both the base FPRA and in specific risk-informed applications. Assumptions are made during the FPRA development to address modeling uncertainties because there is not a single definitive approach. Plant-specific assumptions made for each of the CPS FPRA technical elements are noted in the Uncertainty and Sensitivity Analysis Notebook [6]. The FPRA model uncertainties evaluation considers the modeling uncertainties for the base FPRA by identifying assumptions, determining if those assumptions are related to a source of modeling uncertainty and characterizing that uncertainty, as necessary. Electric Power Research Institute (EPRI) compiled a listing of generic sources of modeling uncertainty to be considered for each FPRA technical element [2], and the evaluation performed for CPS considered each of the generic sources of modeling uncertainty, as well as the plant-specific sources.

Completeness uncertainty addresses scope and level of detail. Uncertainties associated with scope and level of detail are documented in the FPRA but are only considered for their impact on a specific application [6]. No specific issues of PRA completeness have been identified relative to the 50.69 application, based on the results of the FPRA peer reviews.

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

Fire PRA Description	Fire PRA Sources of Uncertainty	Fire PRA Disposition
Analysis Boundary and Partitioning	This task establishes the overall spatial scope of the analysis and provides a framework for organizing the data for the analysis. The partitioning features credited are required to satisfy established industry standards.	Based on a review of the assumptions and potential sources of sources of uncertainly associated with this element, it is concluded that the methodology for the Analysis Boundary and Partitioning task does not introduce any epistemic uncertainties that would affect the 50.69 program.
Fire PRA Component Selection	This task involves the selection of components to be treated in the analysis in the context of initiating events and mitigation. The potential sources of uncertainty include those inherent in the internal events PRA model as that model provides the foundation for the Fire PRA.	The uncertainty associated with this task is related to the identification of all components that should be credited/linked in the FPRA. This source of uncertainty is reduced as a result of multiple overlapping tasks including the MSO expert panel, reviews of FPIE screened initiating events, screened containment penetrations, and screened ISLOCA scenarios. Additional internal reviews of analysis results further reduce the uncertainty associated with this task. Based on a review of the assumptions and potential sources of uncertainty related to this element and the discussion above, it is concluded that the methodology for the Component Selection task does not introduce any epistemic uncertainties that would affect the 50.69 application.

Cable Selection	The selection of cables to be considered in the analysis is identified using industry guidance documents. The overall process is essentially the same as that used to perform the analyses to demonstrate compliance with 10 CFR 50.48.	Additionally, as part of the Fire PRA, some components were conservatively assumed to be failed based on lack of cable data. Components in this category are referred to as Unknown Location (UNL) components because specific cables were not identified for the components. Based on recent Fire PRA updates, the UNL components are mostly limited to Balance of Plant (BOP) systems. Two sensitivity analyses were performed to measure the risk associated with the assumption that these components fail in select fire scenarios. The first sensitivity removed all UNL components from every fire scenario (estimates potential conservatisms) and the second sensitivity evaluated expanded UNL failures in every fire scenario (estimates potential non-conservatisms). The sensitivity analyses are documented in the Uncertainty and Sensitivity Analysis Notebook (Reference [48]). Based on the results of these sensitivity analyses, the UNL methodology does not introduce significant conservatisms into the base FPRA model and is assessed to be appropriate to avoid overly conservative results that mask key risk insights. Given that an informed approach was used to develop the assumed routing, the methodology employed by the FPRA is appropriate. Based on a review of the assumptions and potential sources of uncertainty related to this element it is concluded that the methodology for the Cable Selection task does not introduce any epistemic uncertainties that would affect the 50.69 application.

Fire PRA Description	Fire PRA Sources of Uncertainty	Fire PRA Disposition
Qualitative Screening	Qualitative screening was performed; however, some structures (locations) were eliminated from the global analysis boundary and ignition sources deemed to have no impact on the FPRA (based on industry guidance and criteria) were excluded from the quantification based on qualitative screening criteria. The only criterion subject to uncertainty is the potential for plant trip. However, such locations would not contain any features (equipment or cables identified in the prior two tasks) and consequently are expected to have a low risk contribution.	In the event a structure (location) which could result in a plant trip was incorrectly excluded, its contribution to CDF would be small (with a CCDP commensurate with base risk). Such a location would have a negligible risk contribution to the overall FPRA. Based on a review of the assumptions and potential sources of uncertainty related to this element and the discussion above, it is concluded that the methodology for the Qualitative Screening task does not introduce any epistemic uncertainties that would affect the 50.69 application.
Fire-Induced Risk Model	The internal events PRA model was updated to add fire specific initiating event structure as well as additional system logic. The methodology used is consistent with that used for the internal events PRA model development and was subjected to industry Peer Review. The developed model is applied in such a fashion that all postulated fires are assumed to generate a plant trip. This represents a source of uncertainty, as it is not necessarily clear that fires would result in a trip. In the event the fire results in damage to cables and/or equipment identified in Task 2, the PRA model includes structure to translate them into the appropriate induced initiator.	The identified source of uncertainty could result in the over-estimation of fire risk. In general, the Fire PRA development process would have reviewed significant fire initiating events and performed supplemental assessments to address this possible source of uncertainty. Based on a review of the assumptions and potential sources of uncertainty related to this element and the discussion above, it is concluded that the methodology for the Fire-Induced Risk Model task does not introduce any epistemic uncertainties that would affect the 50.69 application.

Fire PRA Description	Fire PRA Sources of Uncertainty	Fire PRA Disposition
Fire Ignition Frequencies	Fire ignition frequency is an area with inherent uncertainty. Part of this uncertainty arises due to the counting and related partitioning methodology. However, the resulting frequency is not particularly sensitive to changes in ignition source counts. The primary source of uncertainty for this task is associated with the industry generic frequency values used for the Fire PRA. This is because there is no specific treatment for variability among plants along with some significant conservatism in defining the frequencies, and their associated heat release rates.	The FPRA utilized the bin frequencies from NUREG/CR-2169 (Reference 12), which represents the most current approved source for bin frequencies. As such, some of the inherent conservatism associated with bin frequencies from NUREG/CR-6850 was removed. A parametric uncertainty analysis using the Money Carlo method is provided in the Uncertainty and Sensitivity Analysis Notebook (Reference 6). Consensus approaches are employed in the model. Based on a review of the assumptions and potential sources of uncertainty related to this element it is concluded that the methodology for the Fire Ignition Frequency task does not introduce any epistemic uncertainties that would affect the 50.69 application.
Quantitative Screening	Other than screening out potentially risk significant scenarios (ignition sources), this task is not a source of uncertainty.	Quantitative screening criteria was defined for the Clinton Fire PRA as the CDF / LERF contribution of zero, such that all quantified fire scenarios are retained. All of the results were retained in the cumulative CDF / LERF; therefore, no uncertainty was introduced as a result of this task. Based on the discussion above, it is concluded that the methodology for the Quantitative Screening task does not introduce any epistemic uncertainties that would affect the 50.69 application.

Fire PRA Description	Fire PRA Sources of Uncertainty	Fire PRA Disposition
Scoping Fire Modeling	The framework of NUREG/CR-6850 includes two tasks related to fire scenario development. These two tasks are Scoping Fire Modeling and Detailed Fire Modeling. The discussion of uncertainty for both tasks is provided in the discussion for Detailed Fire Modeling.	See discussion for Detailed Fire Modeling.
Detailed Circuit Failure Analysis	The circuit analysis is performed using standard electrical engineering principles. However, the behavior of electrical insulation properties and the response of electrical circuits to fire induced failures is a potential source of uncertainty. This uncertainty is associated with the dynamics of fire and the inability to ascertain the relative timing of circuit failures. The analysis methodology assumes failures would occur in the worst possible configuration, or if multiple circuits are involved, at whatever relative timing is required to cause a bounding worst-case outcome. This results in a skewing of the risk estimates such that they are over-estimated.	Circuit analysis was performed as part of the deterministic post fire safe shutdown analysis. Refinements in the application of the circuit analysis results to the FPRA were performed on a case-by-case basis where the scenario risk quantification was large enough to warrant further detailed analysis. Hot short probabilities and hot short duration probabilities as defined in NUREG-7150, Volume 2, based on actual fire test data, were used in the FPRA. The uncertainty (conservatism) which may remain in the FPRA is associated with scenarios that do not contribute significantly to the overall fire risk. Based on a review of the assumptions and potential sources of uncertainty related to this element and the discussion above, it is concluded that the methodology for the Detailed Circuit Failure Analysis task does not introduce any epistemic uncertainties that would affect the 50.69 application.

Fire PRA Description	Fire PRA Sources of Uncertainty	Fire PRA Disposition
Circuit Failure Model Likelihood Analysis	One of the failure modes for a circuit (cable) given fire induced failure is a hot short. A conditional probability and a hot short duration probability are assigned using industry guidance published in NUREG 7150, Volume 2. The uncertainty values specified in NUREG-7150, Volume 2 are based on fire test data.	The use of hot short failure probability and duration probability is based on fire test data and associated consensus methodology published in NUREG-7150, Volume 2. Based on a review of the assumptions and potential sources of uncertainty related to this element and the discussion above, it is concluded that the methodology for the Circuit Failure Mode Likelihood Analysis task does not introduce any epistemic uncertainties that would affect the 50.69 application.
Detailed Fire Modeling	The application of fire modeling technology is used in the Fire PRA to translate a fire initiating event into a set of consequences (fire-induced failures). The performance of the analysis requires a number of key input parameters. These input parameters include the heat release rate (HRR) for the fire, the growth rate, the damage threshold for the targets, and response of plant staff (detection, fire control, fire suppression). The fire modeling methodology itself is largely empirical in some respects and consequently is another source of uncertainty. For a given set of input parameters, the fire modeling results (temperatures as a function of distance from the fire) are characterized as having some distribution (aleatory uncertainty). The epistemic uncertainty arises from the selection of the input parameters (specifically the HRR and growth	Consensus modeling approach is used for the Detailed Fire Modeling. The methodology for the Detailed Fire Modeling task does not introduce any epistemic uncertainties that would affect the 50.69 application.

Fire PRA Description	Fire PRA Sources of Uncertainty	Fire PRA Disposition
	rate) and how the parameters are related to the fire initiating event. While industry guidance is available, that guidance is derived from laboratory tests and may not necessarily be representative of randomly occurring events. The fire modeling results using these input parameters are used to identify a zone of influence (ZOI) for the fire and cables/equipment within that ZOI are assumed to be damaged. In general, the guidance provided for the treatment of fires is conservative and the application of that guidance retains that conservatism. The resulting risk estimates are also conservative.	
Post-Fire Human Reliability Analysis	The human error probabilities (HEPs) used in the Fire PRA were adjusted to consider the additional challenges that may be present given a fire. The HEPs were obtained using the EPRI HRA Calculator (HRAC) and included the consideration of degradation or loss of necessary cues due to fire. Given the methodology used, the impact of any remaining uncertainties is expected to be small.	The HEPs include the consideration of degradation or loss of necessary cues due to fire. The fire risk importance measures indicate that the results are somewhat sensitive to HRA model and parameter values. The FPRA model HRA is based on industry consensus modeling approaches for its HEP calculations, so this is not considered a significant source of epistemic uncertainty. Further, as directed by NEI 00-04, the fire model human error basic events are increased to their 95th percentile and also decreased to their 5th percentile values as part of the required 50.69 PRA categorization sensitivity cases. These results are capable of driving a component and respective functions HSS and therefore the uncertainty of the PRA modeled HEPs are accounted for in the 50.69 application.

Fire PRA Description	Fire PRA Sources of Uncertainty	Fire PRA Disposition
Seismic-Fire Interactions Assessment	Since this is a qualitative evaluation, there is no quantitative impact with respect to the uncertainty of this task.	The qualitative assessment of seismic induced fires should not be a source of model uncertainty as it is not expected to provide changes to the quantified FPRA model. Based on the discussion of sources of uncertainty and the discussion above, it is concluded that the methodology for the Seismic-Fire Interactions Assessment task does not introduce any epistemic uncertainties that would affect the 50.69 application
Fire Risk Quantification	As the culmination of other tasks, most of the uncertainty associated with quantification has already been addressed. The other source of uncertainty is the selection of the truncation limit. However, the selected truncation was confirmed to be consistent with the requirements of the PRA Standard (Reference [45]).	The selected truncation was confirmed to be consistent with the requirements of the PRA Standard. Based on the discussion of sources of uncertainty and the discussion above, it is concluded that the methodology for the Fire Risk Quantification task does not introduce any epistemic uncertainties that would affect the 50.69 application.
Uncertainty and Sensitivity Analyses	This task does not introduce any new uncertainties. This task is intended to address how the fire risk assessment could be impacted by the various sources of uncertainty.	This task does not introduce any new uncertainties. This task is intended to address how the fire risk assessment could be impacted by the various sources of uncertainty. Based on the discussion of sources of uncertainty and the discussion above, it is concluded that the methodology for the Uncertainty and Sensitivity Analyses task does not introduce any epistemic uncertainties that would affect the 50.69 application.

Fire PRA Description	Fire PRA Sources of Uncertainty	Fire PRA Disposition
Fire PRA Documentation	FPRA Documentation This task does not introduce any new uncertainties to the fire risk.	This task does not introduce any new uncertainties to the fire risk as it outlines documentation requirements. Based on the discussion of sources of uncertainty and the discussion above, it is concluded that the methodology for the Fire PRA Documentation task does not introduce any epistemic uncertainties that would affect the 50.69 application.