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10 CFR 50.90
10 CFR 50.69

November 28, 2018

U.S. Nuclear Regulatory Commission
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Calvert Cliffs Nuclear Power Plant, Units 1 and 2
Renewed Facility Operating License Nos. DPR-53 and DPR-69
NRC Docket Nos. 50-317 and 50-318

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 (Exelon) is requesting an amendment to the license of Calvert Cliffs Nuclear Power Plant (CCNPP), Units 1 and 2.

The proposed amendment would modify the CCNPP 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 CCNPP, Units 1 and 2 Operating Licenses. The categorization process being implemented through this change is consistent with NEI 00-04, "10 CFR 50.69 SSC Categorization Guideline," Revision 0, dated July 2005, which was endorsed by the NRC in Regulatory Guide 1.201, "Guidelines for Categorizing Structures, Systems, and Components in Nuclear Power Plants According to their Safety Significance," Revision 1 dated May 2006. An exception to the NEI 00-04 categorization process described herein is that CCNPP proposes to apply an alternative seismic approach specified in EPRI 3002012988, "Alternative Approaches for Addressing Seismic Risk in 10 CFR 50.69 Risk-Informed Categorization, July 2018" for Tier 1 plants. This approach is a risk-informed graded approach that is demonstrated to produce categorization insights equivalent to a seismic Probabilistic Risk Assessments

(PRAs). For Tier 1 plants, this approach relies on the insights gained from the seismic PRAs examined in EPRI Report 3002012988 along with confirmation that the site Ground Motion Response Spectrum (GMRS) is low. The EPRI approach demonstrates that seismic risk is adequately addressed for Tier 1 sites by the results of the other elements of the 50.69 categorization process.

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.

Though routine maintenance updates have been applied, the Nuclear Regulatory Commission (NRC) has previously reviewed the technical adequacy of the CCNPP PRA models identified in this application for:

- NFPA-805: NRC Safety Evaluation (SE) dated August 30, 2016, ML#16175A359.
- TSTF-505: NRC SE dated October 30, 2018 ML#18270A130

Exelon requests that the NRC utilize the review of the PRA technical adequacy for those applications when performing the review for this application.

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

This letter contains no regulatory commitments.

If you should have any questions regarding this submittal, please contact Enrique Villar at (610) 765-5736.

I declare under penalty of perjury that the foregoing is true and correct. Executed on this 28th day of November 2018.

Respectfully,

David T. Gudger for

James Barstow
Director - Licensing and Regulatory Affairs
Exelon Generation Company, LLC

Enclosure: Evaluation of the Proposed Change

cc: Regional Administrator, NRC Region I
NRC Senior Resident Inspector
NRC Project Manager
D. A. Tancabel, State of Maryland

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LIST OF ATTACHMENTS

1 SUMMARY DESCRIPTION

The proposed amendment modifies the Calvert Cliffs Nuclear Power Plant (CCNPP) licensing basis to allow for the implementation of the provisions of Title 10 of the Code of Federal Regulations (10 CFR), Part 50.69, “Risk-Informed Categorization and Treatment of Structures, Systems and Components for Nuclear Power Reactors.” The provisions of 10 CFR 50.69 allow adjustment of the scope of equipment subject to special treatment controls (e.g., quality assurance, testing, inspection, condition monitoring, assessment, and evaluation). For equipment determined to be of low safety significance, alternative treatment requirements can be implemented in accordance with this regulation. For equipment determined to be of High Safety Significance (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 PRAs, in 2004 the NRC published a new regulation, 10 CFR 50.69. The provisions of 10 CFR 50.69 allow adjustment of the scope of equipment subject to special treatment controls (e.g., quality assurance, testing, inspection, condition monitoring, assessment, and evaluation). For equipment determined to be of low safety significance, alternative treatment requirements can be implemented in accordance with the regulation. For equipment determined to be of high safety significance, requirements will not be changed or will be enhanced. This allows improved focus on equipment that has safety significance resulting in improved plant safety.

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

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

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

2.3 DESCRIPTION OF THE PROPOSED CHANGE

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

> Exelon 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 EPRI alternative approach described in EPRI 3002012988 for seismic risk for Tier 1 plants; 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 is addressed in the following sections.

Though routine maintenance updates have been applied, the NRC has previously reviewed the technical adequacy of the CCNPP Probabilistic Risk Assessment PRA models identified in this application for:

- NFPA-805: NRC SE dated August 30, 2016, ML#16175A359.
- TSTF-505: NRC SE dated October 30, 2018, ML#18270A130

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

3.1.1 Overall Categorization Process

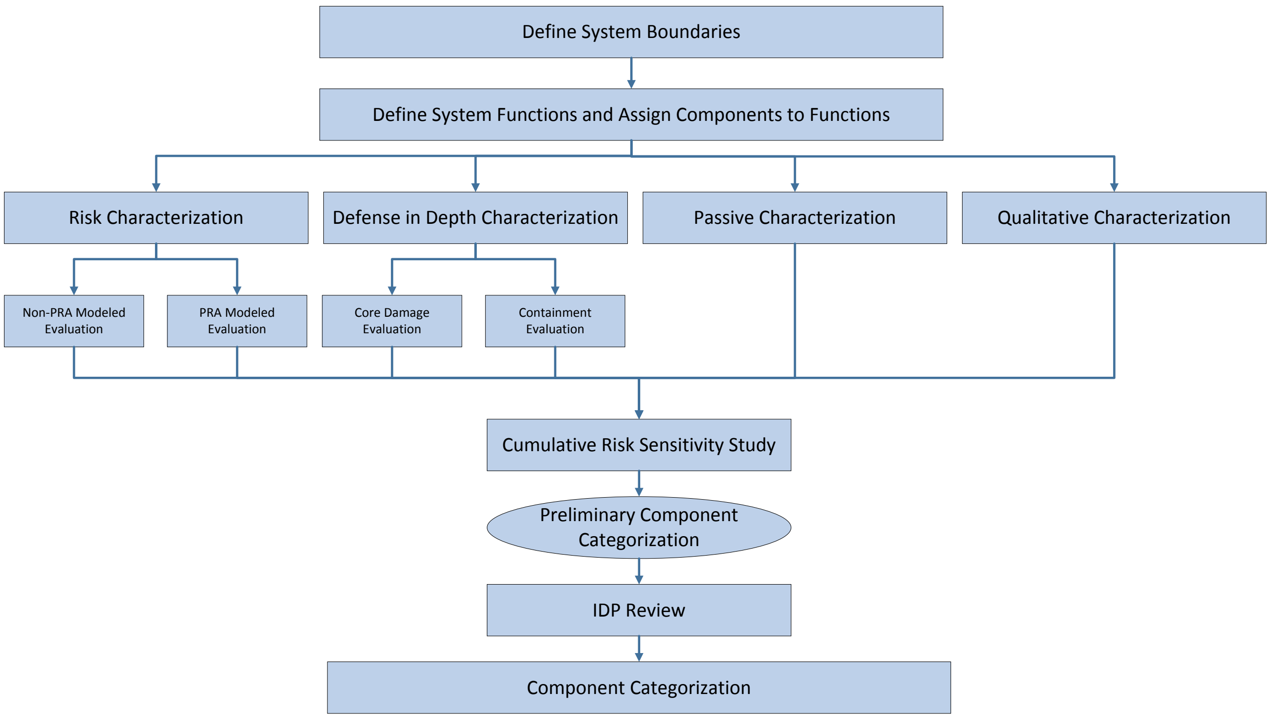
Exelon will implement the risk categorization process in accordance with NEI 00-04, Revision 0, as endorsed by Regulatory Guide (RG) 1.201, "Guidelines for Categorizing Structures, Systems, and Components in Nuclear Power Plants According to their Safety Significance" (Reference 2). NEI 00-04 Section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

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

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, Seismic and Other External Hazards –	Component	Not Allowed	No
	Shutdown – Section 5.5	Function/Component	Not Allowed	No
Defense-in-Depth	Core Damage – Section 6.1	Function/Component	Not Allowed	Yes
	Containment – Section 6.2	Component	Not Allowed	Yes
Qualitative Criteria	Considerations – Section 9.2	Function	Allowable[1]	N/A
Passive	Passive – Section 4	Segment/Component	Not Allowed	No

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

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

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

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

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

- The 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 Exelon 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 3) 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, Exelon will not take credit for alternate means unless the alternate means are proceduralized and included in Licensed Operator training.

- CCNPP proposes to apply an alternative seismic approach to those listed in NEI 00-04 Sections 1.5 and 5.3. This approach is specified in EPRI 3002012988 (Reference 4) 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, accepted by NRC for TSTF 505 dated October 30, 2018. ML#18270A130 (Refer to Attachment 2).

- Fire Risks: Fire PRA model, accepted by the NRC for NFPA 805; NRC SE dated August 30, 2016, ML# 16175A359; also accepted by NRC for TSTF-505, dated October 30, 2018.ML# 18270A130 (Refer to Attachment 2).

- Seismic Risks: EPRI Alternative Approach in EPRI 3002012988 (Reference 4) for Tier 1 plants.

- Other External Risks (e.g., tornados, external floods): Using to the IPEEE screening process as approved by NRC SE dated June 8, 2001, (TAC Nos. M83603 and M83604). 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 5), 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 6 (ML090930246) consistent with the related Safety Evaluation (SE) issued by the Office of Nuclear Reactor Regulation.

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

The use of this method was previously approved to be used for a 10 CFR 50.69 application by NRC in the final Safety Evaluation for Vogtle dated December 17, 2014 (Reference 3). 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 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 for passive categorization is acceptable and appropriate for use at CCNPP for 10 CFR 50.69 SSC categorization.

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

The following sections demonstrate that the quality and level of detail of the processes used in categorization of SSCs are adequate. The PRA models described below have been peer reviewed and there are no PRA upgrades that have not been peer reviewed. The PRA models credited in this request are the same PRA models credited in the NFPA-805 application and related NRC Safety Evaluation Report dated August 30, 2016, (ML#16175A359) and the TSTF-505 application dated February 25, 2016 (ML#16060A223) with routine maintenance updates applied.

3.2.1 Internal Events and Internal Flooding

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

3.2.2 Fire Hazards

The CCNPP 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 Exelon risk management process ensures that the PRA model used in this application reflects the as-built and as-operated plant for each of the CCNPP units. 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 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 CCNPP seismic

hazard assessment, CCNPP proposes to use a risk informed graded approach that meets the requirements of 10 CFR 50.69 (b)(2) as an alternative to those listed in NEI 00-04 sections 1.5 and 5.3. This approach is specified in Reference 4, where CCNPP meets the 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 3002012988 applies to the Tier 1 sites in its entirety except for the sections 2.3 (Tier 2 sites), 2.4 (Tier 3 sites), Appendix A (seismic correlation), and Appendix B (criteria for capacity-based screening).

This approach is a risk-informed graded approach that is demonstrated to produce categorization insights equivalent to a seismic PRA. For Tier 1 plants, this approach relies on the insights gained from the seismic PRAs examined in Reference 4 along with confirmation that the site GMRS is low. Reference 4 demonstrates that seismic risk is adequately addressed for Tier 1 sites by the results of the other elements of the 50.69 categorization process.

As an example, the 50.69 categorization process as defined in NEI 00-04 includes an Integral Assessment that weighs the hazard-specific relative importance of a component (e.g., internal events, internal fire, seismic) by the fraction of the total Core Damage Frequency (CDF) contributed by that hazard. The risk from an external hazard can be reduced from the default condition of HSS if the integral assessment meet 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.

EPRI 3002012988 recommends a risk-informed graded approach for addressing the seismic hazard in the 50.69 categorization process. There are a number of seismic fragility fundamental concepts that support a graded approach and there are important characteristics about the comparison of the seismic design basis (represented by the SSE) to the site-specific seismic hazard (represented by the GMRS) that support the selected thresholds between the three evaluation Tiers in the EPRI report. The coupling of these concepts with the categorization process in NEI 00-04 are the key elements of the approach defined in EPRI 3002012998 for identifying unique seismic insights.

The seismic fragility of a 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 7) 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 CCNPP, there is no need to perform more detailed evaluations to demonstrate the inherent seismic capacities documented in industry sources such as Reference 7. 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 hazards at CCNPP.

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 3002012988.

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 50.69 Integral Assessment if the equipment is HSS only due to seismic considerations.

Test cases described in Section 3 of Reference 4 showed that it would be unusual even for moderate hazard plants to exhibit any unique seismic insights, including due to correlated failures. 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 CCNPP, 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 3002012988 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 CCNPP seismic hazard changes to medium risk (i.e., Tier 2) at some future time, CCNPP 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 CCNPP meets the Tier 1 site criteria.

In response to the NRC 50.54(f) letter associated with post-Fukushima recommendations (Reference 8), CCNPP submitted a seismic hazard screening report (Reference 9) to the NRC. The maximum GMRS value for CCNPP in the 1-10 Hz range meets the Tier 1 criterion of approximately 0.2g in Reference 4. The CCNPP SSE and GMRS curves from the seismic hazard and screening response in Reference 9 are shown in Figure 1 of Attachment 4. The NRC's staff assessment of the CCNPP seismic hazard and screening response is documented in Reference 10. In section 3.4 of Reference 10 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 CCNPP site.

Section 1.1.3 of Reference 4 cites various post-Fukushima seismic reviews performed for the U.S. fleet of nuclear power plants. For CCNPP, 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 9, 10)
2. NTTF Recommendation 2.1 spent fuel pool assessment (References 11, 12)
3. NTTF Recommendation 2.3 seismic walkdowns (References 13 through 17)
4. NTTF Recommendation 4.2 seismic mitigation strategy assessment (S-MSA) (References 18, 19)

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

5. NTTF Recommendation 2.1 seismic expedited seismic evaluation process (ESEP) (References 20, 21)
6. NTTF Recommendation 2.1 seismic high frequency evaluation (References 22, 23)

Based on the above, the Summary/Conclusion/Recommendation from Section 2.2.3 of Reference 4 applies to CCNPP, i.e., CCNPP is a Tier 1 plant for which the GMRS is very low such that unique seismic categorization insights are expected to be minimal. As discussed in Reference 4, the likelihood of identifying a unique seismic insight that would cause an SSC to be designated HSS is very low. Therefore, with little to no anticipated unique seismic insights, the 50.69 categorization process using the Full Power Internal Events (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 CCNPP Units 1 and 2 per a plant-specific evaluation in accordance with GL 88-20 (Reference 24) and updated to use the criteria in ASME PRA Standard RA-Sa-2009. Attachment 4 provides a summary of the external hazards screening results. Attachment 5 provides a summary of the progressive screening approach for external hazards.

3.2.5 Low Power & Shutdown

Consistent with NEI 00-04, the CCNPP 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 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 Exelon risk management process ensures that the applicable PRA model(s) used in this application continues to reflect the as-built and as-operated plant for each of the CCNPP units. The process delineates the responsibilities and guidelines for updating the PRA models, and includes criteria for both regularly scheduled and interim PRA model updates. The process includes provisions for monitoring potential areas affecting the PRA models (e.g., due to changes in the plant, errors or limitations identified in the model, and industry operational experience) for assessing the risk impact of unincorporated changes, and for controlling the model and associated computer files. The process will assess the impact of these changes on the plant PRA model in a timely manner but no longer than once every two refueling outages. If there is a significant impact on the PRA model, the SSC categorization will be re-evaluated.

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

The detailed process of identifying, characterizing and qualitative screening of model uncertainties is found in Section 5.3 of NUREG-1855 (Reference 25) and Section 3.1.1 of EPRI TR-1016737 (Reference 26). 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 CCNPP 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 CCNPP 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 CCNPP PRA model specific assumptions or sources of uncertainty.

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

The PRA models described in Section 3.2 have been assessed against RG 1.200, "An Approach for Determining the Technical Adequacy of Probabilistic Risk Assessment Results for Risk-Informed Activities," Revision 2 (Reference 27) 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 June 2010 (Reference 28). A focused-scope review was conducted in January 2017 on an internal flood PRA model pipe rupture frequency calculation upgrade (Reference 29).

The Fire PRA (FPRA) model was subject to a self-assessment and a full-scope peer review conducted in January 2012 (Reference 30). The NRC evaluated resolution of the FPRA peer review findings and issued its safety evaluation in August 2016 (Reference 31) as part of the transition of CCNPP to incorporate a new fire protection licensing basis in accordance with 10 CFR 50.48(c).

The technical adequacy of the Internal Events PRA (including the 2010 peer review and self-assessment results) and the FPRA (including the January 2012 peer review and self-assessment results) have been previously reviewed by the NRC in the CCNPP TSTF-505 (RICT) Submittal (Reference 32) and in the NFPA-805 Submittal (Reference 31).

A finding closure review was conducted on the internal events PRA model in January 2017. Closed findings were reviewed and closed using the process documented in (draft-proposed) Appendix X to NEI 05-04, NEI 07-12, and NEI 12-13,"Close-out of Facts and Observations" (F&Os) (Reference 33) as accepted by NRC in the letter dated May 3, 2017 (ML17079A427) (Reference 34). The draft-proposed guidance used during the peer review was consistent with the final letter dated May 3, 2017 (ML17079A427). The results of this review have been documented and are available for NRC audit.

Attachment 3 provides a summary of the two open findings remaining after the CCNPP internal event and internal flooding model finding closure review. All of the fire peer review findings have been resolved and accepted by the NRC per its RICT and NFPA-805 safety evaluations (References 32 and 31, respectively).

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

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

The CCNPP 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 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.

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.

4 REGULATORY EVALUATION

4.1 APPLICABLE REGULATORY REQUIREMENTS/CRITERIA

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

- The regulations at 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, U.S. Nuclear Regulatory Commission, March 2009.

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

4.2 NO SIGNIFICANT HAZARDS CONSIDERATION ANALYSIS

Exelon 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.

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

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

    Response: No.

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

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

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

   Response: No.

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

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

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

   Response: No.

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

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

Based on the above, Exelon 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 amendments 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," Revision 1, May 2006.

3. 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.

4. EPRI 3002012988, Alternative Approaches for Addressing Seismic Risk in 10CFR50.69 Risk-Informed Categorization, July 2018.

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

6. ANO SE 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.

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

8. 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).

9. Constellation Energy Nuclear Group, LLC, (CEUS Sites), 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 Near-Term Task Force Review, Calvert Cliffs Nuclear Power Plant, Units 1 and 2, Renewed Facility Operating License Nos. DPR-53 and DPR-69, Docket Nos. 50-317 and 50-318, R.E. Ginna Nuclear Power Plant, Renewed Facility Operating License No. DPR-18, Docket No. 50-244, Nine Mile Point Nuclear Station, Units 1 and 2,

Renewed Facility Operating License Nos. DPR-63 and NPF-69, Docket Nos. 50-220 and 50-410, March 31, 2014 (ML14099A196).

10. U.S. Nuclear Regulatory Commission, Calvert Cliffs Nuclear Power Plant, Units 1 and 2 – Staff Assessment of Information Provided Pursuant to Title 10 of the Code of Federal Regulations Part 50, Section 50.54(F), Seismic Hazard Reevaluations for Recommendation 2.1 of the Near-Term Task Force Review of Insights from the Fukushima Dai-ichi Accident (TAC Nos. MF3970 and MF3971), July 8, 2015 (ML15153A073).

11. Exelon Generation Company, LLC, Spent Fuel Pool Evaluation Supplemental Report, Response to NRC Request for Information Pursuant to 10 CFR 50.54(f) Regarding Recommendation 2.1 of the Near-Term Task Force Review of Insights from the Fukushima Dai-ichi Accident, Calvert Cliffs Nuclear Power Plant, Units 1 and 2, Renewed Facility Operating License Nos. DPR-53 and DPR-69, NRC Docket Nos. 50-317 and 50-318, RS-16-123, August 31, 2016 (ML16244A320).

12. Exelon Generation Company, LLC, Calvert Cliffs Nuclear Power Plant, Units. 1 and 2 - Staff Review of Spent Fuel Pool Evaluation Associated with Reevaluated Seismic Hazard Implementing Near-Term Task Force Recommendation 2.1 and Staff Closure of Activities Associated with Recommendation 2.1 (CAC NOS. MF3970 and MF3971), October 12, 2016 (ML16281A491).

13. Constellation Energy Nuclear Group, LLC, Response to 10 CFR 50.54(f) Request for Information, Recommendation 2.3, Seismic, Calvert Cliffs Nuclear Power Plant, Unit 1, Renewed Facility Operating License No. DPR-53, Docket No. 50-317, November 27, 2012 (ML12349A281).

14. Constellation Energy Nuclear Group, LLC, Response to 10 CFR 50.54(f) Request for Information, Recommendation 2.3, Seismic, Calvert Cliffs Nuclear Power Plant, Unit 2, Renewed Facility Operating License No. DPR-69, Docket No. 50-318, November 27, 2012 (ML12339A349).

15. Constellation Energy Nuclear Group, LLC, Supplemental Response to 10 CFR 50.54(f) Request for Information, Recommendation 2.3, Seismic, Calvert Cliffs Nuclear Power Plant, Unit 2, Renewed Facility Operating License No. DPR-69, Docket No. 50-318, June 28, 2013 (ML13193A150).

16. Constellation Energy Nuclear Group, LLC, Response to Request for Additional Information Associated with Near-Term Task Force Recommendation 2.3, Seismic Walkdowns, Calvert Cliffs Nuclear Power Plant, Units a and 2, Renewed Facility Operating License Nos. DPR-53 and DPR-69, Docket Nos. 50-317 and 50-318, Nine Mile Point Nuclear Station, Units 1 and 2, Renewed Facility Operating License Nos. DPR-63 and NPF-69, Docket Nos. 50-220 and 50-410, R.E. Ginna Nuclear Power Plant, Renewed Facility Operating License No. DPR-18, Docket No. 50-244, December 2, 2013 (ML13346A011).

17. U.S. Nuclear Regulatory Commission, Calvert Cliffs Nuclear Power Plant, Unit Nos. 1 and 2, R. E. Ginna Nuclear Power Plant, and Nine Mile Point Nuclear Station, Unit Nos. 1 and 2 - Staff Assessment of Seismic Walkdown Reports Supporting Implementation of Near-Term Task Force Recommendation 2.3 Related to Fukushima Dai-ichi Nuclear Power Plant Accident (TAC Nos. MF0104 and MF0105, MF0127, MF0145 and MF0146), June 2, 2014 (ML14134A133).

18. Exelon Generation Company, LLC, Seismic Mitigating Strategies Assessment (MSA) Report for the Reevaluated Seismic Hazard Information - NEI 12-06, Appendix H, Revision 4, H.4.4 Path 4: GMRS < 2xSSE, Calvert Cliffs Nuclear Power Plant, Units 1 and 2, Renewed Facility Operating License Nos. DPR-53 and DPR-69, NRC Docket Nos. 50-317 and 50-318, August 31, 2017 (ML17243A018).

19. U.S. Nuclear Regulatory Commission, Calvert Cliffs Nuclear Power Plant, Units 1 And 2 – Staff Review of Mitigating Strategies Assessment Report of the Impact of The Re-Evaluated Seismic Hazard Developed in Response to the March 12, 2012, 50.54(f) Letter (CAC Nos. MF7812 and MF7813; EPID L-2016-JLD-0006), February 7, 2018 (ML18033A209).

20. Exelon Generation Company, LLC, Expedited Seismic Evaluation Process Report (CEUS Sites), response to NRC Request for Information Pursuant to 10CFR 50.54(f) Regarding Near-Term Task Force Review of Insights from the Fukushima Dai-ichi Accident, Calvert Cliffs Nuclear Power Plant, Units 1 and 2, Renewed Facility Operating License Nos. DPR-53 and DPR-69, NRC Docket Nos. 50-317 and 50-318, FLL-14-036, December 17, 2014 (ML14365A138).

21. U.S. Nuclear Regulatory Commission, Calvert Cliffs Nuclear Power Plant, Units 1 and 2 – Staff Review of Interim Evaluation Associated with Reevaluated Seismic Hazard Implementing Near-Term Task Force Recommendation 2.1 (TAC NOS. MF5231 and MF5232), September 23, 2015 (ML15238A429).

22. Exelon Generation Company, LLC, High Frequency Supplement to Seismic Hazard Screening Report, Response to NRC Request for Information Pursuant to 10 CFR 50.54(f) Regarding Recommendation 2.1 of the Near-Term Task Force Review of Insights from the Fukushima Dai-ichi Accident, Calvert Cliffs Nuclear Power Plant, Units 1 and 2, 10 CFR 50.54(f) Renewed Facility Operating License Nos. DPR-53 and DPR-69, NRC Docket Nos. 50-317 and 50-318, RS-15-286, December 4, 2015 (ML15338A002).

23. U.S. Nuclear Regulatory Commission, Staff Review of High Frequency Confirmation Associated with Reevaluated Seismic Hazard in Response to March 12, 2012 50.54(F) Request for Information, To the Power Reactor Licensees on the Enclosed List (includes Calvert Cliffs Nuclear Power Plant, Units 1 and 2), February 18, 2016 (ML15364A544).

24. Generic Letter 88-20, "Individual Plant Examination of External Events (IPEEE) for Severe Accident Vulnerabilities - 10 CFR 50.54(f), Supplement 4," USNRC, June 1991.

25. NUREG-1855, Guidance on the Treatment of Uncertainties Associated with PRAs in Risk-Informed Decision Making, Revision 1, March 2017.

26. EPRI TR-1016737, Treatment of Parameter and Model Uncertainty for Probabilistic Risk Assessments, December 2008.

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

28. LTR-RAM-II-10-055, "RG 1.200 PRA Peer Review Against the ASME PRA Standard Requirements for The Calvert Cliffs Nuclear Power Plant Probabilistic Risk Assessment," Westinghouse, November 23, 2010.

29. CCNPP PRA Finding Level Fact And Observation Technical Review & Focused-Scope Peer Review, Report No. 032299-RPT-001 Revision 0.

30. LTR-RAM-12-02, "Fire PRA Peer Review of the Calvert Cliffs Nuclear Power Plant Fire Probabilistic Risk Assessment Against the Fire PRA Standard Supporting Requirements for Section 4 of the ASME/ANS Standard," Westinghouse, April 24, 2012.

31. Calvert Cliffs Nuclear Power Plant, Unit Nos. 1 and 2: Issuance of Amendments Regarding Transition To A Risk-informed, Performance-Based Fire Protection Program In Accordance With 10 CFR 50.48(c) (CAC NOS. MF2993 AND MF2994), August 30, 2016 (ML16175A359).

32. Calvert Cliffs Nuclear Power Plant, Units 1 and 2 – Issuance of Amendment Nos. 326 and 304 to Add Risk-Informed Completion Time Program (EPID L-2016-LLA-0001), October 30, 2018

33. NEI Letter to USNRC, "Final Revision of Appendix X to NEI 05-04/07-12/12-13, Close-Out of Facts and Observations (F&Os)," February 21, 2017, Accession Number ML17086A431.

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

35. "CCNPP Aircraft Hazard Analysis," RAN: 97-034B1, Revision 5, February 22, 2002.

36. Letter from Exelon Generation to NRC, "Response to March 12, 2012, Request for information Enclosure 2, Recommendation 2.1, Flooding, Required Response 2, Flood Hazard Reevaluation Report," September 23, 2015 (ML15272A311)

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38. CNPP Individual Plant Examination of External Events, RAN-97-031

39. Letter from J.A. Spina (Constellation Energy) to NRC, "Calvert Cliffs Nuclear Power Plant; Unit Nos. 1 & 2; Docket Nos. 50-317 & 50-318; Independent Spent Fuel Storage Installation; Docket No. 72-8; Revision to Hazards Analysis Related to Liquefied Natural Gas Plant Operations at Cove Point," February 20, 2008 [ADAMS ML080560423].

40. Letter from Mr. R. E. Denton (BGE) to Document Control Desk (NRC), dated June 7, 1993, "Liquefied Natural Gas Hazards Analysis."

41. CCNPP Unit 3 Combined Licensed Application, Part 2: Final Safety Analysis Report, Revision 10, September 2014

42. NPP Updated Final Safety Analysis Report (UFSAR), Section 5.3.1, Revision 47

43. CA-PRA-026, Rev. 1, "Calvert Cliffs [Internal Events PRA] Uncertainty Assessment Notebook," February 2018.

44. CO-UNC-001, Rev. 1, "Calvert Cliffs [Fire PRA] Uncertainty and Sensitivity Analysis Notebook," August 2013.

45. NUREG/CR-6850 (also EPRI 1011989), "Fire PRA Methodology for Nuclear Power Facilities," September 2005, with Supplement 1 (EPRI 1019259), September 2010.

46. "Nuclear Power Plant Fire Ignition Frequency and Non-Suppression Probability Estimation Using the Updated Fire Events Database, United States Fire Event Experience Through 2009," NUREG-2169/EPRI 3002002936, U.S. NRC and Electric Power Research Institute, January 2015.

47. "Refining And Characterizing Heat Release Rates From Electrical Enclosures During Fire (RACHELLE-FIRE), Volume 1: Peak Heat Release Rates and Effect of Obstructed Plume," NUREG-2178 Vol. 1/ EPRI 3002005578, U.S. NRC and Electric Power Research Institute, Draft Report for Comment, April 2015.

48. "Joint Assessment of Cable Damage and Quantification of Effects from Fire (JACQUE-FIRE), Volume 2: Expert Elicitation Exercise for Nuclear Power Plant Fire-Induced Electrical Circuit Failure," Final Report, NUREG/CR-7150, Vol. 1, EPRI 3002001989, U.S. NRC and Electric Power Research Institute, May 2014.

Attachment 1: List of Categorization Prerequisites

Exelon 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 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

<table>
<tr><th>Units</th><th>Model</th><th>Baseline CDF</th><th>Baseline LERF</th><th>Comments</th></tr>
<tr><td colspan="5">Internal Events / Internal Flood Model</td></tr>
<tr><td>1</td><td rowspan="2">CCNPP Unit 1 and Unit 2 Version CA015A

Peer Reviewed Against RG 1.200 R2 in November 2010</td><td>9.5E-06</td><td>1.2E-06</td><td rowspan="2">NRC reviewed for risk-informed completion times (ML 18270A130)</td></tr>
<tr><td>2</td><td>9.6E-06</td><td>1.2E-06</td></tr>
<tr><td colspan="5">Internal Fire Model</td></tr>
<tr><td>1</td><td rowspan="2">CCNPP
Unit 1 and Unit 2 Version CC014A-W-CRU

Peer Reviewed Against RG 1.200 R2 in January 2012</td><td>4.2E-05</td><td>3.2E-06</td><td rowspan="2">NRC reviewed for risk-informed completion times (ML 18270A130) and NFPA-805 (ML 16175A359).</td></tr>
<tr><td>2</td><td>4.0E-05</td><td>3.4E-06</td></tr>
</table>

Attachment 3: Disposition and Resolution of Open Peer Review Findings

F&O ID	SR	Topic	Finding/Observation	Status	Disposition for 50.69
1-18 (from 2010 peer review)	IFSO-A4 IFEV-A7	Internal Flooding	Examined Internal Flooding Notebook (C0-IF-001, Rev. 1) Section 3.3 and 5.3. Consideration of human-induced mechanisms as potential flood sources not clear. Regarding human-induced impacts on the flood frequency, Section 5.3 of the IF report states that they were included, but their inclusion should be better documented or referenced from IF (e.g., a sample calculation showing human contribution would be helpful) (This F&O originated from SR IFSO-A4 Capability Category 1-3 is MET	Partially Resolved with Open Documentation (IFSO-A4) Resolved (IFEV-A7)	This is an internal flood documentation finding. There is no impact to the 50.69 LAR because the considerations are included in the model. The Calvert Cliffs PRA Internal Flood notebook has been updated to address the finding.

F&O ID	SR	Topic	Finding/Observation	Status	Disposition for 50.69
IFFS-01 (finding from 2017 focused scope peer review)	IFEV-B1 IFEV-B2	Internal Flooding	The Internal Flood Notebook assembled in performing the upgrade to the internal flood PRA was judged not to satisfy the requirement that it document the internal flood-induced initiating events in a manner that facilitates PRA applications, upgrades, and peer review. Essential inputs to the calculation of flood frequencies associated with pipe ruptures is distributed among a variety files, some of which are not formally stored with other PRA information. Reconstructing the initiating frequencies was impossible without the assistance of members of the IFPRA team. NOT MET Capability Category I-III.	Open	This is an internal flood documentation finding. There is no impact to the 50.69 LAR because the internal flood-induced initiating event frequencies are included in the model and were verified by the peer review team. Additional documentation will be added to the Calvert Cliffs PRA Internal Flood notebook to allow for easier verification in the future. The finding has been captured in the PRA configuration control database.

Attachment 4: External Hazards Screening

External Hazard	Screening Result		
	Screened? (Y/N)	**Screening Criterion (Note a)**	**Comment**
Aircraft Impact	Y	C2	A post-IPEEE analysis of aircraft crash rates for CCNPP was performed in 2002. The analysis determined that the aircraft crash rate into safety related structures is less than 10-6/yr (Reference 35).
Avalanche	Y	C3	Not applicable to the site because of climate and topography.
Biological Event	Y	C3 C5	Slow developing hazard, can be detected and managed. Plant programs are in place to periodically inspect and clean.
Coastal Erosion	Y	C5	Shoreline erosion is discussed in Section 2.4.4 of the UFSAR. Shoreline recession along the site is due mainly to wave erosion, particularly storm waves, undercutting the cliff. Shoreline recession has occurred at an average rate of about 0.7 inches/yr. Approximately 3700 lineal ft of shore protection has been placed in front of the plant area. Shore protection consists of onsite material placed in front of the cliffs and faced with filter cloth and layered riprap.

External Hazard	Screening Result		
	Screened? (Y/N)	Screening Criterion (Note a)	Comment
Drought	Y	C1 C5	Plant design eliminates drought as a concern. In addition, this event is slowly developing.
External Flooding	Y	C1 PS2	The external flooding hazard at the site was recently updated as a result of the post-Fukushima Flood Hazard Reevaluation Request (FHRR) (Reference 36). On September 23, 2015, the FHRR was revised to account for more site-specific parameters. As a result, the revised hazards are now considered bounded by the current design basis at the plant. In the Staff Assessment of the Focused Evaluation dated December 20, 2017 (Reference 37), NRC concludes that the plant only relies on permanently installed passive flood barriers, mainly plant grade, to prevent any impacts to SSCs from external flooding hazards.

External Hazard	Screening Result		
	Screened? (Y/N)	**Screening Criterion (Note a)**	**Comment**
Extreme Wind or Tornado	Y	C4 PS2 PS4	The combined hurricane and tornado risk for both units was evaluated to be less than 1E-6/yr and 1E-7/yr for CDF and LERF respectively. NEI 00-04 requires that, as part of the external hazard screening, an evaluation be conducted to determine if there are components that participate in screened scenarios and whose failure would result in an unscreened scenario. Such SSCs are required to be high safety significant in the categorization process. A list of credited SSCs has been developed and will be used during categorization of applicable systems.
Fog	Y	C1 C4	Fog and mist may increase the frequency of accidents involving aircraft, ships, or vehicles. This weather condition is included implicitly in the accident rate data for these Transportation Accidents.

External Hazard	**Screening Result**		
	Screened? (Y/N)	**Screening Criterion (Note a)**	**Comment**
Forest or Range Fire	Y	C3 C4	Fires due to transportation accidents, industrial facilities, and pipelines are discussed in their respective sections in this attachment. For forest fires, fire is unlikely to propagate to the site, because the site is cleared for several hundred feet (Reference 38). Fire in the vicinity of the plant could potentially result in a loss of offsite power (LOOP). The potential LOOP due to a forest fire is accounted for in the internal events PRA.
Frost	Y	C1 C4	Included implicitly in weather-related LOOP.
Hail	Y	C1 C4	Building design for high wind and missiles is bounding. Included implicitly in weather-related LOOP initiator.
High Summer Temperature	Y	C1 C5	Plant AC ventilation is designed for extreme heat load. This event is sufficiently slowly developing that there is a long time to respond to air temperature rise.
High Tide, Lake Level, or River Stage	Y	C1 C4	Impact covered in External Flooding Hazard.

External Hazard	Screening Result		
	Screened? (Y/N)	**Screening Criterion (Note a)**	**Comment**
Hurricane	Y	C4	Impact covered in the extreme wind or tornado hazard, and in the external flooding hazard.
Ice Cover	Y	C1 C4	Plant is designed for freezing temperatures. Ice cover is implicitly included in plant response to LOOP events.

External Hazard	Screening Result		
	Screened? (Y/N)	Screening Criterion (Note a)	Comment
Industrial or Military Facility Accident	Y	C1 C3 PS4	The Dominion Cove Point Liquid Natural Gas (LNG) Terminal is located approximately 3½ miles from CCNPP. An analysis was performed in 2006 as part of a proposed expansion of the terminal and its operations (Reference 39). Previous studies (e.g., Reference 40) concluded that the risk to CCNPP was negligible. The updated study estimates the risk of LNG operations contained within the area, including LNG ships enroute, berthing of ships and cargo transfer, storage and processing at the onshore facility, and pipeline export. The risk of fatality at CCNPP and the risk of physical damage to the plant was estimated at significantly less than 10-6/yr. There are no other substantial industrial or military facilities within 5 miles of CCNPP (Reference 41). Explosive hazard impacts and control room habitability impacts meet the 1975 SRP requirements (RGs 1.91 and 1.78).
Internal Flooding	N	None	The CCNPP 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	None	The CCNPP internal events PRA includes evaluation of risk from internal fire events.
Landslide	Y	C3	Not applicable to the site because of topography.
Lightning	Y	C1	Lightning strikes causing loss of offsite power or turbine trip are contributors to the initiating event frequencies for these events. However, other causes are also included. The impacts are no greater than already modeled in the internal events PRA.
Low Lake Level or River Stage	Y	C3	Not applicable to the site because of location.
Low Winter Temperature	Y	C1 C5	Extended freezing temperatures are rare, the plant is designed for such events, and their impacts are slow to develop.
Meteorite or Satellite Impact	Y	PS4	Likelihood of a large meteorite, large enough to cause significant plant damage, is very low.

External Hazard	Screening Result		
	Screened? (Y/N)	Screening Criterion (Note a)	Comment
Pipeline Accident	Y	C2 C3	A pipeline from the Dominion Cove Point LNG Terminal runs through the center of Calvert County and passes about 2 miles from CCNPP. The study (Reference 39) performed for the LNG terminal includes hazards from this pipeline. Based on the study, the pipeline accident hazard to the site is much less than 10-6/yr. There are no other known pipelines in the vicinity of CCNPP.
Release of Chemicals in Onsite Storage	Y	C1 PS2	A separate hazard analysis was performed as part of the IPEEE (Reference 38). The analysis screened and/or evaluated the probability of a release which could result in a loss of Control Room habitability, incapacitation of operators, and damage to vital equipment and subsequent off site exposure levels exceeding 10CFR100 limits. It was concluded that no chemicals stored onsite posed a significant hazard to Control Room operability or plant equipment. There are no challenges presented to the CCNPP site from chemicals stored onsite.
River Diversion	Y	C3	Not applicable to the site because of location.

External Hazard	Screening Result		
	Screened? (Y/N)	Screening Criterion (Note a)	Comment
Sand or Dust Storm	Y	C1 C3 C5	This event is generally not applicable due to plant location. Potential impacts covered under loss of offsite power events.
Seiche	Y	C3	Not applicable to the site because of location.
Seismic Activity	Y	C1 C2	See Section 3.2.3 and Figure A4-1 in this Attachment.
Snow	Y	C1 C4	The event damage potential is less than other events for which the plant is designed. Potential flooding impacts covered under external flooding.
Soil Shrink-Swell Consolidation	Y	C1 C5	The potential for this hazard is low at the site, the plant design considers this hazard, and the hazard is slowly developing and can be mitigated.
Storm Surge	Y	C4	Included under External Flooding.
Toxic Gas	Y	C4	Toxic gas covered under release of chemicals in onsite storage, industrial or military facility accident, and transportation accident.

External Hazard	Screening Result		
	Screened? (Y/N)	Screening Criterion (Note a)	Comment
Transportation Accident	Y	C1 C2 C3 C4	Analysis of accidents on transportation routes (other than airways) in the vicinity of CCNPP was performed in the IPEEE (Reference 38). Additionally, a more recent study performed for Calvert Cliffs Unit 3 (Reference 41) showed that transportation accidents (other than aircraft) represented a negligible risk to CCNPP. Aircraft accidents are discussed in a previous part of this attachment.
Tsunami	Y	C3	Not applicable to the site because of location.

External Hazard	Screening Result		
	Screened? (Y/N)	**Screening Criterion (Note a)**	**Comment**
Turbine-Generated Missiles	Y	C1 C2 PS2	CCNPP Turbine Missile risk is managed using the NRC- preferred method of maintaining P1 (turbine missile generation probability) at a low value. For unfavorably oriented turbines, such as at CCNPP, the specified value is less than 1 x 10-5/yr. This ensures the risk to the public from turbine-missile-induced core damage and release is well less than 1E-06. The value of P1 for each unit is calculated by each turbine vendor and maintained by performance of vendor recommended maintenance and testing of turbine valves and overspeed controls. P1 values are well less than 1E-05 for each unit. CCNPP UFSAR (Reference 42) Section 5.3 provides further details and references. There are no challenges presented to the CCNPP site from turbine generated missiles.
Volcanic Activity	Y	C3	Not applicable to the site because of location.
Waves	Y	C3 C4	Waves associated with adjacent large bodies of water are not applicable to the site. Waves associated with external flooding are covered under that hazard.

<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 colspan="4">Note a – See Attachment 5 for descriptions of the screening criteria.</td></tr>
</table>

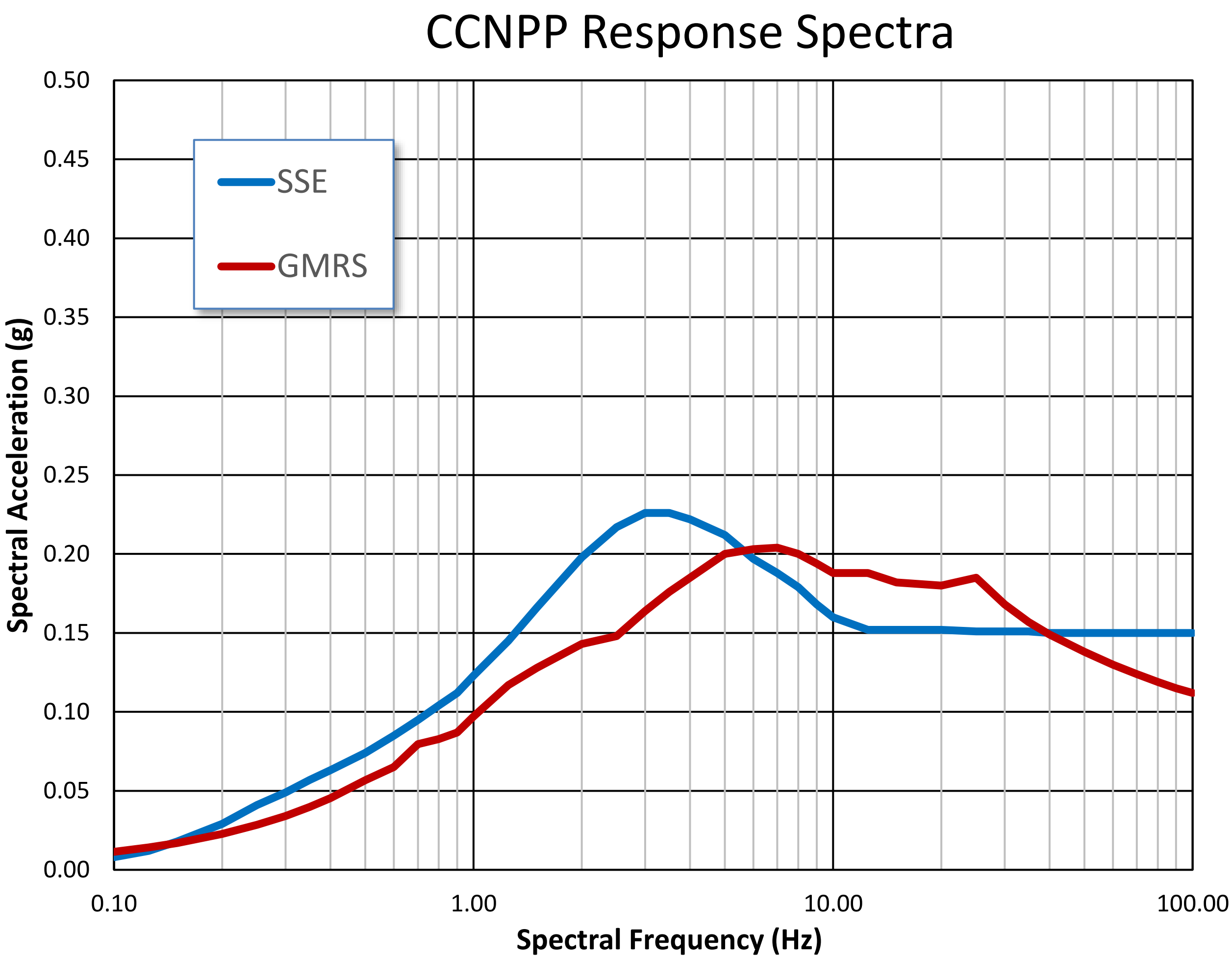

Figure 1: GMRS and SSE Response Spectra for CCNPP
(From Reference 9, Figures 2.4-1 and 3.1-1)

Attachment 5: Progressive Screening Approach for Addressing External Hazards

Event Analysis	Criterion	Source	Comments
Initial Preliminary Screening	C1. Event damage potential is < events for which plant is designed.	NUREG/CR-2300 and ASME/ANS Standard RA-Sa-2009	
	C2. Event has lower mean frequency and no worse consequences than other events analyzed.	NUREG/CR-2300 and ASME/ANS Standard RA-Sa-2009	
	C3. Event cannot occur close enough to the plant to affect it.	NUREG/CR-2300 and ASME/ANS Standard RA-Sa-2009	
	C4. Event is included in the definition of another event.	NUREG/CR-2300 and ASME/ANS Standard RA-Sa-2009	Not used to screen. Used only to include within another event.
	C5. Event develops slowly, allowing adequate time to eliminate or mitigate the threat.	ASME/ANS Standard RA-Sa-2009	
Progressive Screening	PS1. Design basis hazard cannot cause a core damage accident.	ASME/ANS Standard RA-Sa-2009	
	PS2. Design basis for the event meets the criteria in the NRC 1975 Standard Review Plan (SRP).	NUREG-1407 and ASME/ANS Standard RA-Sa-2009	
	PS3. Design basis event mean frequency is < 1E-5/y and the mean conditional core damage probability is < 0.1.	NUREG-1407 as modified in ASME/ANS Standard RA-Sa-2009	
	PS4. Bounding mean CDF is < 1E-6/y.	NUREG-1407 and ASME/ANS Standard RA-Sa-2009	

Event Analysis	Criterion	Source	Comments
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

Reference 43 and Reference 44 document sources of PRA modeling uncertainty. They identify assumptions and determine if those assumptions are related to sources of model uncertainty and characterize that uncertainty, as necessary. The uncertainties in Reference 43 and Reference 44 were reviewed for this application. The tables below contain the identified items for evaluation as potential key sources of uncertainty for this 10 CFR 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 Uncertainty and Assumptions	IE/IF PRA 50.69 Impact	IE/IF PRA Model Sensitivity and Disposition (50.69)
Uncertainties associated with the assumptions and method of calculation of HEPs for the Human Reliability Analysis (HRA) may introduce uncertainty. Detailed evaluations of HEPs are performed for the risk significant human failure events (HFEs) using industry consensus methods. Mean values are used for the modeled HEPs. Uncertainty associated with the mean values can have an impact on CDF and LERF results.	Potentially all SSCs evaluated during 50.69 categorization	Sensitivity cases for the base internal events PRA (use of 5th and 95th percentile value HEPs) show that the results are somewhat sensitive to HRA model and parameter values. Use of 95th percentile HEPs for applications is not considered realistic given the consistent use of a consensus HRA approach. The CCNPP PRA model is based on industry consensus modeling approaches for its HEP calculations, so this is not considered a significant source of epistemic uncertainty. However, as directed by NEI 00-04, internal events 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.

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 the discussion of sources of uncertainly it is concluded that the methodology for the Analysis Boundary and Partitioning task does not introduce any epistemic uncertainties that would require sensitivity treatment. Therefore, the 50.69 calculations are not impacted.
Component Selection	This task involves the selection of components to be treated in the analysis in the context of initiating events and mitigation. The potential sources of uncertainty include those inherent in the internal events PRA model as that model provides the foundation for the FPRA.	In the context of the FPRA, the uncertainty that is unique to the analysis is related to initiating event identification. However, that impact is minimized through use of the PWROG Generic MSO list and the process used to identify and assess potential MSOs. Based on the discussion of sources of uncertainty and the discussion above, it is concluded that the methodology for the Component Selection task does not introduce any epistemic uncertainties that would require sensitivity treatment. Therefore, the 50.69 calculations are not impacted.

Fire PRA Description	**Fire PRA Sources of Uncertainty**	**Fire PRA Disposition**
Cable Selection	The selection of cables to be considered in the analysis is identified using industry guidance documents. The overall process is the same as that used to perform the analyses to demonstrate compliance with 10 CFR 50.48.	Based on the discussion of sources of uncertainty it is concluded that the methodology for the Cable Selection task does not introduce any epistemic uncertainties that would require sensitivity treatment. Therefore, the 50.69 calculations are not impacted.
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 in NUREG/CR-6850) were excluded from the quantification based on qualitative screening criteria. The only criterion subject to uncertainty is the potential for plant trip. However, such locations would not contain any features (equipment or cables identified in the prior two tasks) and consequently are expected to have a low risk contribution.	In the event a structure (location) which could result in a plant trip was incorrectly excluded, its contribution to CDF would be small (with a CCDP commensurate with base risk). Such a location would have a negligible risk contribution to the overall FPRA. Based on the discussion of sources of uncertainty and the discussion above, it is concluded that the methodology for the Qualitative Screening task does not introduce any epistemic uncertainties that would require sensitivity treatment. Therefore, the 50.69 calculations are not impacted.

Fire PRA Description	Fire PRA Sources of Uncertainty	Fire PRA Disposition
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 as 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 FPRA development process would have reviewed significant fire initiating events and performed supplemental assessments to address this possible source of uncertainty. Based on the discussion of sources of uncertainty and the discussion above, it is concluded that the methodology for the Fire-Induced Risk Model task does not introduce any epistemic uncertainties that would require sensitivity treatment. Therefore, the 50.69 calculations are not impacted.

Fire PRA Description	**Fire PRA Sources of Uncertainty**	**Fire PRA Disposition**
Fire Ignition Frequency	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 FPRA. This is because there is no specific treatment for variability among plants along with some significant conservatism in defining the frequencies, and their associated heat release rates. CCNPP currently uses the NUREG/CR-6850 Supplement 1 (Reference 45) ignition frequencies. The fire frequency values are believed to currently be over-estimated. A future model update will address the new ignition frequencies in NUREG-2169 (Reference 46) along with the recently revised heat release rates from NUREG 2178 (Reference 47). This is considered to be part of the normal FPRA model maintenance process.	Based on the discussion of sources of uncertainty, it is concluded that the methodology for the Fire Ignition Frequency task does not introduce any epistemic uncertainties that would require sensitivity treatment. Consensus approaches are employed in the current model and will be employed as appropriate in future model updates. Therefore, the 50.69 calculations are not impacted.

Fire PRA Description	**Fire PRA Sources of Uncertainty**	**Fire PRA Disposition**
Quantitative Screening	Other than screening out potentially risk significant scenarios (ignition sources), this task is not a source of uncertainty.	The CCNPP FPRA development did not screen out any fire initiating events based on low CDF/LERF contribution. Screening of individual fire ignition sources occurred only if it involved a discrete component and the consequences of the associated fire did not involve failure of any other plant component or feature. Based on the discussion of sources of uncertainty and the discussion above, it is concluded that the methodology for the Quantitative Screening task does not introduce any epistemic uncertainties that would require sensitivity treatment. Therefore, the 50.69 calculations are not impacted.
Scoping Fire Modeling	The framework of NUREG/CR-6850 includes two tasks related to fire scenario development. These two tasks are 8 and 11. The discussion of uncertainty for both tasks is provided in the discussion for Task 11.	Consensus modeling approach is used for the Detailed Fire Modeling. It is concluded that the methodology for the Detailed Fire Modeling task does not introduce any epistemic uncertainties that would require sensitivity treatment. Therefore, the 50.69 calculations are not impacted.

Fire PRA Description	Fire PRA Sources of Uncertainty	Fire PRA Disposition
Detailed Circuit Failure Analysis	The circuit analysis is performed using standard electrical engineering principles. However, the behavior of electrical insulation properties and the response of electrical circuits to fire induced failures is a potential source of uncertainty. This uncertainty is associated with the dynamics of fire and the inability to ascertain the relative timing of circuit failures. The analysis methodology assumes failures would occur in the worst possible configuration, or if multiple circuits are involved, at whatever relative timing is required to cause a bounding worst-case outcome. This results in a skewing of the risk estimates such that they are over-estimated.	Circuit analysis was performed as part of the deterministic post fire safe shutdown analysis. Refinements in the application of the circuit analysis results to the FPRA were performed on a case-by-case basis where the scenario risk quantification was large enough to warrant further detailed analysis. Hot short probabilities and hot short duration probabilities as defined in NUREG 7150, Volume 2 (Reference 48), based on actual fire test data, were used in the CCNPP Fire PRA. 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 the discussion of sources of uncertainty 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 require sensitivity treatment. Therefore, the 50.69 calculations are not impacted.

Fire PRA Description	**Fire PRA Sources of Uncertainty**	**Fire PRA Disposition**
Circuit Failure Mode Likelihood Analysis	One of the failure modes for a circuit (cable) given fire induced failure is a hot short. A conditional probability and a hot short duration probability are assigned using industry guidance published in NUREG 7150, Volume 2 (Reference 48). 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 the discussion of sources of uncertainty 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 require sensitivity treatment. Therefore, the 50.69 calculations are not impacted.
Detailed Fire Modeling	The application of fire modeling technology is used in the FPRA to translate a fire initiating event into a set of consequences (fire induced failures). The performance of the analysis requires a number of key input parameters. These input parameters include the heat release rate (HRR) for the fire, the growth rate, the damage threshold for the targets, and response of plant staff (detection, fire control, fire suppression). The fire modeling methodology itself is largely empirical in some respects and consequently is another source of uncertainty. For a given set of input parameters, the fire modeling results (temperatures as a	Consensus modeling approach is used for the Detailed Fire Modeling. It is concluded that the methodology for the Detailed Fire Modeling task does not introduce any epistemic uncertainties that would require sensitivity treatment. Therefore, the 50.69 calculations are not impacted.

Fire PRA Description	Fire PRA Sources of Uncertainty	Fire PRA Disposition
	function of distance from the fire) are characterized as having some distribution (aleatory uncertainty). The epistemic uncertainty arises from the selection of the input parameters (specifically the HRR and growth rate) and how the parameters are related to the fire initiating event. While industry guidance is available, that guidance is derived from laboratory tests and may not necessarily be representative of randomly occurring events. The fire modeling results using these input parameters are used to identify a zone of influence (ZOI) for the fire and cables/equipment within that ZOI are assumed to be damaged. In general, the guidance provided for the treatment of fires is conservative and the application of that guidance retains that conservatism. The resulting risk estimates are also conservative.	

Fire PRA Description	**Fire PRA Sources of Uncertainty**	**Fire PRA Disposition**
Post-Fire Human Reliability Analysis	The human error probabilities used in the FPRA were adjusted to consider the additional challenges that may be present given a fire. The human error probabilities were obtained using the EPRI 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 human error probabilities were obtained using the EPRI HRAC and included the consideration of degradation or loss of necessary cues due to fire. The impact of any remaining uncertainties is expected to be small. Further, as directed by NEI 00-04, 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 require sensitivity treatment. Therefore, the 50.69 calculations are not impacted.
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.	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 require sensitivity treatment. Therefore, the 50.69 calculations are not impacted

Fire PRA Description	**Fire PRA Sources of Uncertainty**	**Fire PRA Disposition**
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 require sensitivity treatment. Therefore, the 50.69 calculations are not impacted.
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 FPRA documentation task does not introduce any epistemic uncertainties that would require sensitivity treatment. Therefore, the 50.69 calculations are not impacted.