UNITED STATES
NUCLEAR REGULATORY COMMISSION
WASHINGTON, D.C. 20555-0001

March 20, 2019

Mr. Bryan C. Hanson
Senior Vice President
Exelon Generation Company, LLC
President and Chief Nuclear
Officer Exelon Nuclear
4300 Winfield Road
Warrenville, IL 60555

SUBJECT: CALVERT CLIFFS NUCLEAR POWER PLANT, UNITS 1 AND 2 – AUDIT PLAN IN SUPPORT OF REVIEW OF LICENSE AMENDMENT REQUEST TO ALLOW USE OF RISK-INFORMED CATEGORIZATION AND TREATMENT OF STRUCTURES, SYSTEMS, AND COMPONENTS (EPID L-2018-LLA-0482)

Dear Mr. Hanson:

By letter dated November 28, 2018 (Agencywide Documents Access and Management System (ADAMS) Accession No. ML18333A022), Exelon Generation Company, LLC (Exelon, the licensee) submitted a license amendment request (LAR) for Calvert Cliffs Nuclear Power Plant, Units 1 and 2 (Calvert Cliffs). The proposed amendments would modify the licensing basis to allow for the implementation of the provisions of Title 10 of the *Code of Federal Regulations* Part 50.69, "Risk-Informed Categorization and Treatment of Structures, Systems and Components for Nuclear Power Reactors." The proposed amendments include an exception to the U.S. Nuclear Regulatory Commission (NRC)-endorsed categorization process in Nuclear Energy Institute (NEI) 00-04, Revision 0, "10 CFR 50.69 SSC Categorization Guideline, dated July 2005 (ADAMS Accession No. ML052910035), to apply an alternative seismic approach for the seismic hazard specified in Electric Power Research Institute Report 3002012988, "Alternative Approaches for Addressing Seismic Risk in 10 CFR 50.69 Risk-Informed Categorization," dated July 2018.

The NRC staff has reviewed Exelon's LAR and determined that a regulatory audit of the alternate seismic approach would assist in the timely completion of the LAR review. The NRC staff will conduct a regulatory audit to support its review of the LAR in accordance with the enclosed audit plan. The staff notes that the scope of their audit information needs related to the technical acceptability of the probabilistic risk assessments used to develop insights to support the licensee's proposed approach, and the mapping of components in different probabilistic risk assessment models can be affected based on the response to questions related to the consideration of seismic events during categorization in the proposed approach.

The audit will be conducted in two parts. The first part of the audit will be conducted from March 26, 2019, to March 28, 2019, at Exelon's office located at 200 Exelon Way, Kennett Square, Pennsylvania, between 8:00 a.m. and 4:00 p.m. each day. The second part will be held via telephone conference between April 8, 2019 and April 12, 2019 and will be limited to the first item listed in Section IV of the enclosed audit plan. A regulatory audit is a planned activity that includes the examination and evaluation of primarily non-docketed information. The audit will be conducted to increase the NRC staff's understanding of the LAR and identify information that will require docketing to support the NRC staff's regulatory finding.

The logistics and scope of the audit were discussed with your staff on March 18, 2019. The audit plan is enclosed.

If you have any questions, please contact me by telephone at 301-415-2871 or e-mail to Michael.Marshall@nrc.gov.

Sincerely,

Michael L. Marshall, Jr.
Senior Project Manager
Plant Licensing Branch I
Division of Operating Reactor Licensing
Office of Nuclear Reactor Regulation

Docket Nos. 50-317 and 50-318

Enclosure:
Audit Plan

cc: Listserv

AUDIT PLAN

REGARDING RISK-INFORMED CATEGORIZATION AND TREATMENT OF

STRUCTURES, SYSTEMS, AND COMPONENTS

EXELON GENERATION COMPANY, LLC

CALVERT CLIFFS NUCLEAR POWER PLANT, UNITS 1 AND 2

DOCKET NOS. 50-317 AND 50-318

I. BACKGROUND

By letter dated November 28, 2018 (Agencywide Documents Access and Management System (ADAMS) Accession No. ML18333A022), Exelon Generation Company, LLC (Exelon, the licensee) submitted a license amendment request (LAR) for Calvert Cliffs Nuclear Power Plant, Units 1 and 2 (Calvert Cliffs). The proposed amendments would modify the licensing basis to allow for the implementation of the provisions of Title 10 of the *Code of Federal Regulations* (10 CFR) Part 50.69, "Risk-Informed Categorization and Treatment of Structures, Systems and Components for Nuclear Power Reactors." The proposed amendments include an exception to the U.S. Nuclear Regulatory Commission (NRC)-endorsed categorization process in Nuclear Energy Institute (NEI) 00-04, Revision 0, "10 CFR 50.69 SSC Categorization Guideline," dated July 2005 (ADAMS Accession No. ML052910035), to apply an alternative approach for the seismic hazard specified in Electric Power Research Institute (EPRI) Report 3002012988, "Alternative Approaches for Addressing Seismic Risk in 10 CFR 50.69 Risk-Informed Categorization," dated July 2018.[1] The NRC staff has reviewed Exelon's submittal and determined that a regulatory audit of the Calvert Cliffs alternate seismic approach would assist in the timely completion of the subject LAR review process.

II. REGULATORY AUDIT BASES

Regulatory Guide (RG) 1.201, Revision 1, "Guidelines for Categorizing Structures, Systems, and Components in Nuclear Power Plants According to Their Safety Significance," dated May 2006 (ADAMS Accession No. ML061090627), endorses, with clarifications and qualifications, NEI 00-04, Revision 0, as one acceptable method for use in complying with the requirements in 10 CFR 50.69. The NEI 00-04 guidance describes in detail a process for determining the safety significance of structures, systems, and components (SSCs) and categorizing them into the four risk-informed safety class (RISC) categories defined in 10 CFR 50.69. This categorization process uses an integrated decision-making process, incorporating both risk and traditional engineering insights. The NEI 00-04 guidance allows licensees to implement different approaches, depending on the scope of their probabilistic risk assessment (PRA). The proposed amendments include an exception to the NRC-endorsed categorization process in NEI 00-04 to apply an alternative seismic approach specified in EPRI 3002012988.

[1]All references to the EPRI report in this document refer to this report. The same report is cited as Reference 4 in the enclosure to the licensee's submittal dated November 28, 2018, and is publicly-available free of cost online at https://www.epri.com/#/pages/product/000000003002012988/?lang=en-US.

The regulation in 10 CFR 50.69(b)(2)(ii) provides the requirements to describe 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 other systematic evaluation techniques used to evaluate severe accident vulnerabilities) are adequate for the categorization of SSCs. The regulation in 10 CFR 50.69(c)(ii) requires the categorization process to determine SSC functional importance using an integrated, systematic process for addressing initiating events (internal and external), SSCs, and plant operating modes, including those not modeled in the plant-specific PRA. The functions to be identified and considered include design bases functions and functions credited for mitigation and prevention of severe accidents. All aspects of the integrated, systematic process used to characterize SSC importance must reasonably reflect the current plant configuration and operating practices, and applicable plant and industry operational experience. Finally, the regulation in 10 CFR 50.69(e) requires periodic updates to the licensee's PRA and SSC categorization.

RG 1.200, Revision 2, "An Approach for Determining the Technical Adequacy of Probabilistic Risk Assessment Results for Risk-Informed Activities," March 2009 (ADAMS Accession No. ML090410014), describes an approach for determining whether the base PRA in total, or the parts that are used to support an application, is acceptable for use in regulatory decision-making for light-water reactors. RG 1.200 endorses, with staff clarifications and qualifications, the 2009 version of the American Society of Mechanical Engineers (ASME)/American Nuclear Society (ANS) PRA Standard (ASME/ANS RA-Sa-2009).

The audit will be performed consistent with NRC Office of Nuclear Reactor Regulation Office Instruction LIC-111, "Regulatory Audits," dated December 29, 2008 (ADAMS Accession No. ML082900195). An audit was determined to be the most efficient approach toward a timely resolution of issues associated with this LAR review, since the staff will have an opportunity to minimize the potential for multiple rounds of requests for additional information (RAIs) and ensure no unnecessary burden will be imposed by requiring the licensee to address issues that are no longer necessary to make a safety determination. Upon completion of this audit, the staff is expected to develop and issue RAIs, as needed, to allow completion of the LAR review, and the licensee will be expected to provide the necessary information on the docket. The final RAIs will be issued after the audit.

III. PURPOSE AND SCOPE

The purpose of the audit is to gain a more detailed understanding of the basis and implementation of the licensee's alternate seismic approach in its proposed categorization process and to gain more information relevant to the review of the subject LAR. Specifically, the NRC staff will be given an overview of the licensee's alternate approach to consider the seismic risk. The licensee will discuss the identified issues related to the technical and regulatory bases of its proposed approach for this application and the unique technical aspects associated with using the approach in its categorization process.

The areas of focus for the regulatory audit are the information contained in the licensee's submittal; the enclosed audit information needs; and all associated and relevant supporting documentations, including methodology, process information, calculations, etc. The relevant supporting documents are identified later in this audit plan.

IV. INFORMATION AND OTHER MATERIAL NECESSARY FOR THE REGULATORY AUDIT

The following documentation should be available to the audit team:

1. For each of Plants A through D in the EPRI report, a listing of SSCs identified as LSS based on the corresponding internal events and Fire PRAs which either have a design basis function during seismic events or functions credited for mitigation and prevention of severe accidents caused by seismic events.
2. Seismic Walkdown Report for Calvert Cliffs in response to Near Term Task Force Recommendation 2.3.
3. Calvert Cliffs Individual Plant Examination of External Events (IPEEE) submittal.

In addition to making the above documents available to the NRC staff, the licensee should be prepared to discuss each of the audit information needs included with this plan. The audit team will not remove non-docketed information from the audit site.

V. AUDIT TEAM

The members of the audit team are anticipated to be:

- Michael Marshall, Project Manager, NRC
- Mehdi Reisi Fard, Team Leader, PRA, NRC
- Shilp Vasavada, Reliability and Risk Analyst, PRA, NRC
- David Heeszel, Geophysicist, NRC (participation by telephone)

VI. LOGISTICS

The audit will be conducted in two parts. The first part of the audit will be conducted from March 26, 2019, to March 28, 2019, at Exelon's office located at 200 Exelon Way, Kennett Square, Pennsylvania, between 8:00 a.m. and 4:00 p.m. each day. The second part will be held via telephone conference between April 8, 2019 and April 12, 2019 and will be limited to the first item listed in Section IV of this audit plan. An entrance briefing will be held at the beginning of the first part of the audit, and an exit briefing will be held at the end of the second part of the audit. The NRC project manager will coordinate any changes to the audit schedule and location with the licensee.

VII. SPECIAL REQUESTS

The NRC staff would like access to the following equipment and services:

- Telephone with a speaker or speaker phone
- Enclosed conference room (or comparable space) with a table, chairs, and white board
- Breakout room for NRC staff discussions
- Projector and screen
- Wireless internet access (if available in the work space)

VIII. DELIVERABLES

An audit summary, which may be public, will be prepared within 90 days of the completion of the audit. If the NRC staff identifies information during the audit that is needed to support its regulatory decision on the submittal, the staff will issue RAIs to the licensee after the audit.

IX. AUDIT INFORMATION NEEDS

Consideration of Seismic Events in Proposed Calvert Cliffs Tier 1 Approach

1. Section 3.2.4, "Other External Hazards," of the enclosure to the LAR states that, "All external hazards, except for seismic, were screened..." However, Attachment 4 of the enclosure to the LAR includes a tabulation of the screening results for various external hazards and states that "seismic activity" is screened (i.e., "Y" under the "Screened? Y/N" column). Section 3.2.3, "Seismic Hazard," of the enclosure to the LAR states that little to no unique seismic insights are anticipated for Calvert Cliffs and that the seismic risk is adequately addressed by other elements of the proposed categorization process. The staff notes that justifying the screening of a hazard, as done in Attachment 4 of the enclosure to the LAR, has the same purpose (i.e., that use of an external hazard PRA does not provide unique insights for the application, and therefore, is not required for categorization). However, the screened hazards are included in the IDP decision-making through the presentation and consideration of the results of qualitative evaluations (e.g., the process in Figure 5-6 of NEI 00-04). It is unclear whether such evaluations will be performed for the seismic hazard similar to other hazards that are screened out.

   a. If the licensee's intent is to screen the seismic hazard and seismic risk from any future considerations in the categorization process, justify exclusion of qualitative evaluations used for other screened hazards (e.g., the process in Figure 5-6 of NEI 00-04). The discussion should include any potential interaction of the process in Figure 5-6 of NEI 00-04 with the proposed alternative seismic approach and the technical, as well as regulatory issues, discussed in questions 2, 3, 4, and 5.

   b. If the exclusion of qualitative considerations of the seismic hazard and seismic risk cannot be justified, discuss and justify qualitative considerations that would provide the IDP a means to consider potential impacts of seismic events in the categorization process. The qualitative consideration should address the technical, as well as regulatory issues, discussed in questions 2, 3, 4, and 5.

2. The regulation in 10 CFR 50.69(c)(ii) requires the categorization process to determine SSC functional importance using an *integrated, systematic* process for addressing initiating events (internal and *external*), SSCs, and plant operating modes, including those not modeled in the plant-specific PRA. The functions to be identified and considered include *design bases functions* and functions credited for *mitigation and prevention of severe accidents*. All aspects of the integrated, systematic process used to characterize SSC importance must reasonably reflect the current plant configuration and operating practices, and applicable plant and industry operational experience (*italics* used to add emphasis).

The proposed approach is based on risk insights from four case studies. Those case studies compared the high safety significance (HSS) SSCs determined based on a seismic PRA (SPRA) against HSS SSCs determined from other PRA models used for categorization.

Each of the cases studies included a corresponding full power internal events (FPIE) PRA but only two of the four case studies used information from a fire PRA. The guidance in NEI 00-04, as endorsed in RG 1.201, maintains the HSS categorization for SSCs identified as such from the FPIE PRA and allows the use of the integrated importance measure determination, as well as the IDP for SSCs categorized as "candidate" HSS from the fire PRA.

Information in the EPRI report shows the SSCs identified as HSS from SPRAs that overlapped with SSCs identified as HSS from corresponding fire PRAs. Such SSCs represent about 21 percent of the HSS fragility groups for Plant C, which is a substantial amount, especially given the limited use of fire PRAs in the development of the insights. In addition, one fragility group was identified as HSS only from fire PRA for Plant A. The staff notes that a fragility group includes multiple components, and therefore, at the SSC level, HSS categorization from fire PRAs contributes even more. The staff notes that Plants B and D do not have a fire PRA, and Plant D used its Appendix R safe shutdown list for comparison.

Based on the use of fire PRAs in the development of insights, SSCs identified as HSS from fire PRA that are not HSS from FPIE and have mitigation functions for seismically-induced events, have a noticeable contribution to the insights that SPRAs identify limited, unique HSS SSCs. The insights (i.e., HSS categorization of the SSC) gained from the case studies that use fire PRA may be changed by the subsequent categorization of such SSCs as LSS, following the endorsed guidance. Therefore, it is unclear how the licensee's categorization process adequately considers the safety significance of SSCs identified as HSS only by Calvert Cliffs fire PRA and has design-basis functions during seismic events or functions credited for mitigation and prevention of severe accidents caused by seismic events.

Considering that (1) the proposed approach relies, in part, on the fire PRA model to identify risk-significant SSCs that would have been identified by an SPRA, and (2) the proposed categorization process appears to allow changing HSS SSCs identified by fire PRA without any consideration of seismic risk, describe, with justification, how the Calvert Cliffs categorization process considers HSS SSCs identified by Calvert Cliffs fire PRA that are not HSS from FPIE but have design-basis functions during seismic events or functions credited for mitigation and prevention of severe accidents caused by seismic events.

3. The regulation in 10 CFR 50.69(c)(ii) requires the categorization process to determine SSC functional importance using an integrated, systematic process for addressing initiating events (internal and external), SSCs, and plant operating modes. Two acceptable approaches to consider the risk of seismic events in the categorization process include the seismic margins analysis and SPRAs. Since all SSCs that are part of the safe shutdown equipment list from the seismic margins analysis are assigned as HSS, such SSCs will be treated according to the guidance in Section 9.2.1, "Review of Safety-Significant Functions/SSCs," of NEI 00-04 by the IDP, and separate

consideration for LSS categorization is unnecessary. With respect to the use of SPRAs, the guidance in Sections 6, 7, and 9 of NEI 00-04 does not explicitly cite the impact of seismic events, or seismic failure modes (e.g., anchorage failure, brittle fracture, seismic interactions, and correlated failures), as part of the corresponding considerations. Nevertheless, the SSCs identified as LSS from an SPRA would undergo the qualitative considerations (e.g., the defense-in-depth considerations in Section 6 of NEI 00-04), as well as the consideration by the IDP discussed in Section 9.2.2, "Review of Safety Related Low Safety-Significant Functions/SSCs," of NEI 00-04.

The Statements of Consideration accompanying the publication of 10 CFR 50.69 state that (*italics* used to add emphasis), "The cornerstone of 50.69 is the establishment of a robust, risk-informed *categorization process* that provides high confidence that the safety significance of SSCs is correctly determined *considering all relevant information*," and "The process is structured to *ensure that all relevant information pertaining to SSC safety significance is considered by a panel*...and that the assembled information is considered in a manner that *ensures the Commission's criteria for risk-informed applications are satisfied*..." The Statements of Consideration further state that (*Italics* used to add emphasis), "The IDP decision process can be viewed as an extension of the previous process for determining SSC safety classification...in that it is *making use of relevant risk information that was not considered or not available* when the SSCs were initially classified."

However, the proposed approach appears to rely solely on insights from PRA importance measures and excludes the consideration of seismic events during the qualitative considerations and IDP deliberations. It is unclear how the proposed approach meets the intent of the 10 CFR 50.69 rule as clarified in the Statements of Consideration, and how the impact of SSCs for mitigation of seismically-induced events, as well as seismic failure modes of such SSCs, will be adequately considered without inclusion in qualitative determinations. Such qualitative determinations include engineering evaluations (e.g., Section 7 of NEI 00-04), IDP deliberations (e.g., Section 9.2.2 of NEI 00-04), defense-in-depth considerations in Section 6 of NEI 00-04, categorization related procedures, and IDP training.

The staff notes that the information available to the licensee such as the system selection and walkdowns performed in response to Near Term Task Force Recommendation 2.3 plant response characteristics from, and walkdowns for, the licensee's IPEEE SPRA, and design-basis seismic requirements from the Updated Final Safety Analysis Report (UFSAR) can provide input about mitigation of seismically-induced events, as well as seismic failure modes for qualitative determinations.

Justify how the proposed approach meets the intent of the 10 CFR 50.69 rule given that the proposed approach seems to exclude consideration of the impact of SSCs on the mitigation of seismically-induced events, as well as seismic failure modes of such SSCs, as part of the qualitative and IDP evaluations cited above. Alternately, propose a mechanism to include such consideration in the licensee's implementation of the 10 CFR 50.69 program.

4. Section 3.6.3, "Relays," of the EPRI report discusses the insights from the case studies related to relays and the implementation of the insights in the proposed approach. The discussion cites the NEI 00-04 guidance on implicitly modeled

components and states that the function of the relays would need to be evaluated to perform categorization down to the component level. The results for all four case studies in Section 3.2, "Plant A Trial Categorization Evaluation," through Section 3.5, "Plant D Trial Categorization Evaluation," of the EPRI report, identify multiple relays as HSS and describe the 'mapping' of those relays to a component in the FPIE PRA. Further, the discussion appears to focus entirely on relays in emergency power systems. The staff notes that the development of SPRAs involves an identification (e.g., by circuit analysis) of relays that impact the function of components credited for mitigation of seismically-induced initiating events. Therefore, the fact that a relay is modeled in an SPRA is sufficient to demonstrate that the relay impacts the function of the corresponding system (i.e., the evaluation has already been performed as part of SPRA development). It is unclear whether the IDP can change the categorization (HSS to LSS) for components that are implicitly modeled (such as relays in the proposed approach) in the Calvert Cliffs categorization process.

a. Section 3.6.3 of the EPRI report appears to only discuss relays associated with emergency power systems. The SPRAs used for the case studies provide insights rather than definitive conclusions. The NRC staff notes that the insights discussed in Section 3.6.3 of the EPRI report are that (1) relays are separately modeled in SPRAs, (2) relays are categorized as HSS based on the SPRAs, (3) the impact of relays can be captured through SSCs modeled in the FPIE through the guidance for implicitly modeled component, and (4) the guidance for functional evaluation at the component level can be used to categorize the (implicitly modeled) relays. Therefore, the insights are not specific to the emergency power system relays and are applicable to relays for any SSCs where the seismically-induced failure modes of the relays can negatively impact the function of that SSC. The NRC staff notes that the case studies represent a subset of plants and that plant configurations, mitigation features, and operations vary.

   Justify why the insights in Section 3.6.3 of the EPRI report are adequate and should not be extended to all relays impacting the ability of SSCs to perform their function, considering that the insights from the case studies have applicability beyond only the emergency power system relays.

b. Discuss the parts of the NRC-endorsed guidance in NEI 00-04 that are used by the IDP to categorize implicitly modeled components based on their function (such as relays). Include clarification of whether the IDP can change the categorization of components that are not modeled in the PRA (such as implicitly modeled SSCs) citing the relevant guidance that allows or prevents the IDP from such categorization decisions. If the IDP can change the categorization of implicitly modeled components and the licensee intends to use the corresponding guidance, justify the rationale for allowing an assignment of LSS to relays that would have been categorized as HSS from an SPRA and would have contributed to the integral importance measure of the 'mapped' SSC (also known as the 'supercomponent').

5. The discussions in Section 3.3, "Plant B Trial Categorization Evaluation," and Section 3.5 for Plants B and D, respectively, state that those two plants have diverse and flexible coping strategies (FLEX) equipment explicitly modeled in their PRAs, including their SPRA, and that FLEX equipment has been identified as HSS based on

the SPRA. Some of the HSS FLEX equipment would be categorized as HSS from the FPIE PRA but several examples exist in the case studies of Plants B and D where FLEX equipment is identified as HSS uniquely from SPRA. Based on information available to the NRC staff, Plant A's PRAs appear to also include credit for FLEX equipment, and such equipment would be categorized as HSS, based on Plant A's SPRA. The NRC staff notes that the insights derived from the case studies related to FLEX equipment include that (1) three of the four case studies model FLEX equipment, and (2) such equipment is identified as HSS either uniquely from the SPRA or from the SPRA and FPIE PRA both. The insights are not unexpected due to the fact that the purpose of FLEX equipment is to support mitigation of sequences primarily caused by external events.

Section 3.6.4, "FLEX Components," of the EPRI report discusses the insights from the case studies related to FLEX equipment modeled in the PRAs and the implementation of the insights in the proposed approach. The discussion on alternative treatments in Section 3.6.4 of the EPRI report states that meeting PRA technical acceptability guidance in RG 1.200, Revision 2, ensures that the performance assumed in the PRA for FLEX equipment is consistent with plant practices.

The NRC staff notes that the guidance in RG 1.200, Revision 2, ensures that SSC reliability data used in PRAs reflects the as built, as-operated plants, and the guidance is not used to maintain assumed performance of such SSCs. The regulations in 10 CFR 50.69(e) require the licensee to "ensure that the treatment of RISC-2 SSCs is consistent with the assumed performance in the PRA," which is related to performance monitoring and 'feedback loop' subsequent to the potential categorization of FLEX equipment into RISC-2. Three of the four case studies identified FLEX equipment as HSS uniquely from the SPRA. The identification would contribute to the potential categorization of those SSCs into RISC-2 based on the integrated importance measure determination. Sole reliance on PRA technical acceptability of the internal events and/or fire PRAs, without consideration of the insights related to the contribution of FLEX equipment to mitigation during seismically-induced events from the case studies, would result in FLEX equipment being categorized into RISC-4, as opposed for RISC-2, and the 'feedback loop' required by 10 CFR 50.69 (e) not being applicable for such equipment. Because the case studies in the EPRI document identified FLEX equipment as HSS uniquely from SPRAs, exclusion of the insights related to FLEX equipment has the potential of resulting in incorrect categorization of those components. Therefore, it is unclear to the staff how insights from the EPRI report related to identification of FLEX SSCs that are HSS from SPRA or other PRAs will be considered, qualitatively and/or quantitatively, in the Calvert Cliffs proposed approach.

a. Discuss how the Calvert Cliffs categorization process considers the insights from the EPRI report related to the potential risk significance of FLEX equipment for mitigating seismically-induced events.

b. If FLEX equipment is not explicitly evaluated qualitatively and/or quantitatively in the context of mitigation of seismically-induced events, justify exclusion of such consideration, given the insights in the EPRI report that FLEX equipment can be HSS based on the contribution from seismic risk, and therefore, will be part of the 'feedback loop' required by 10 CFR 50.69 (e).

Calvert Cliffs Seismic Risk, Performance Monitoring, and Seismic Hazard Change

6. Section 3.2.3 of the enclosure to the LAR provides discussion on the "basis for establishing Tier 1 criteria." The preceding discussion in that section is related to the seismic fragility of SSCs in comparison to the design basis at a plant such as the licensee. The basis for the Calvert Cliffs Tier 1 criteria is derived from consideration of the margin in the inherent seismic capacity, the seismic design basis, and that the seismic risk (core damage frequency (CDF)/large early release frequency (LERF) will be low.

   The risk insights derived from the four case studies in the EPRI report are based on SPRAs that represent a probabilistic treatment of the response of SSCs during a seismic event and the plant behavior therefrom. Therefore, the insights derived from the case studies are not limited to the design-basis seismic event. Section 3.2.3 of the enclosure to the LAR states that the proposed approach would "produce categorization insights equivalent to a seismic PRA." However, the Calvert Cliffs Tier 1 classification is based on Calvert Cliffs' capability to respond to the design-basis seismic event. Evaluation of response to design-basis seismic events may not produce insights equivalent to those from the case studies (i.e., SPRAs) that consider the full range of hazard. The comparison of the ground motion response spectrum and safe shutdown earthquake only demonstrate that the seismic risk may not have changed compared to the plant's original design basis. The licensee's SPRA submitted in response to IPEEE included correlated failures that appeared as the dominant risk contributors.

   The staff's safety evaluation dated October 30, 2018 (ADAMS Accession No. ML18270A130) for the licensee's request to adopt Technical Specifications Task Force (TSTF) Traveler TSTF-505 states that the site-specific seismic risk estimate presented by the licensee is 5×10^{-6} per year for CDF and 5×10^{-7} per year for LERF. Those estimates result in an approximately 13 percent contribution to the plant CDF, and an approximately 8 percent contribution to the plant LERF. SSCs can be categorized as HSS based solely on the consideration of seismic risk and with Fussell-Vesely importance of 0.04 and risk achievement worth importance of about 15 (based on the above-mentioned seismic CDF value), which are observed in PRA models. Further, SSCs with Fussell-Vesely and Risk Achievement Worth values less than those cited above may also be determined to be HSS based on the integrated importance measure determination, when combined with the importance measures from other hazard PRAs. The staff notes that the case studies in the EPRI report do not include integrated importance measure determination. The staff notes that the Calvert Cliffs Tier 1 approach does not appear to include any qualitative and/or quantitative considerations of the impact of seismic events. Because the seismic risk at Calvert Cliffs based on docketed information appears to be comparable to that from other hazards, it is unclear whether Calvert Cliffs belongs to Tier 1 (i.e., no consideration of seismic risk in the categorization process). Further, because SPRAs evaluate the full spectrum of the seismic hazard, it is unclear how the proposed approach would produce categorization insights equivalent to an SPRA.

   Considering that (1) the insights in the EPRI report are derived from the full spectrum of the seismic hazard (i.e., not limited to the design basis), while the approach appears to exclude qualitative and/or quantitative consideration of the mitigation capabilities of SSCs during seismically-induced events, as well as seismic failure modes that can prevent an SSC from performing its function (the technical and regulatory issues on these topics are identified in questions 1 through 5), and (2) that the contribution of the quantified seismic risk at Calvert Cliffs based on docketed information appears to be comparable to that from other hazards, justify that the seismic risk is low enough relative to the other hazards such that the categorization results will not be significantly impacted.

7. The regulation in 10 CFR 50.69(c)(1)(iv) requires the categorization process to include evaluations that provide reasonable confidence that for SSCs categorized as RISC-3, any potential increase in CDF and LERF resulting from changes in treatment is small. The regulations in 10 CFR 50.69(e)(2) and (3) require the licensee to monitor the performance of RISC-1 and RISC-2 SSCs and consider the data collected for RISC-3 SSCs and adjust the categorization or treatment processes so that the categorization process and results are maintained valid.

   SSCs that mitigate the impact of seismically-induced events are susceptible to structural failure modes. Such structural failure modes are dependent on several parameters, including the SSC testing and as-built installation, all of which can be impacted by alternative treatments. It is unclear how the licensee's performance monitoring programs, either existing, planned, and/or enhanced, will ensure compliance with the requirements in 10 CFR 50.69(e), as well as any qualitative determinations related to SSC performance during seismic events.

   a. Considering that the approach appears to exclude qualitative and/or quantitative consideration of the mitigation capabilities of SSCs during seismically-induced events, as well as seismic failure modes that can prevent an SSC from performing its function (the technical and regulatory issues on these topics are identified in questions 1 through 5), describe how performance monitoring will be used to ensure that the structural capacity of components continues to maintain compliance with the requirements of 10 CFR 50.69(e) (i.e., monitoring the performance and data collection to ensure categorization remains valid).

   b. If qualitative determinations are included in the proposed Calvert Cliffs Tier 1 approach based on the responses to the technical, as well as regulatory issues raised in questions 1 through 5, describe how performance monitoring will be used to ensure that the inputs for the qualitative determinations continue to remain valid to maintain compliance with the requirements of 10 CFR 50.69(e).

8. Section 2.3, "Description of the Proposed Change," of the enclosure to the LAR proposes a license condition to the renewed operating licenses for Calvert Cliffs. The license condition includes, among other additions, using the "EPRI alternative approach described in EPRI 3002012988 for seismic risk for Tier 1 plants" as one of the categorization processes. The license condition further states that, "Prior NRC approval, under 10 CFR 50.90, ["Application for amendment of license, construction permit, or early site permit,"] is required for a change to the categorization process specified above."

   However, Section 3.2.3, "Seismic Hazards," of the enclosure to the LAR, states that, "In the unlikely event that the Calvert Cliffs seismic hazard changes to medium risk (i.e., Tier 2) at some future time, Calvert Cliffs 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 statement in Section 3.2.3 of the enclosure to the LAR appears to be inconsistent with the proposed license condition because prior NRC approval is the appropriate change control for a change in the categorization process for seismic risk, considering that the submittal (i.e., a licensee-specific LAR) proposes a Calvert Cliffs Tier 1 approach (i.e., plant and application specific).

Explain the intent of the cited statement in Section 3.2.3 of the enclosure to the LAR in the context of the current plant-specific application and resolve the seeming inconsistency between the proposed license condition in Section 2.3 and the cited statement in Section 3.2.3 of the enclosure to the LAR.

Technical Acceptability of PRAs Used for Case Studies

Section 3.2.3 of the enclosure to the LAR states that, "This approach relies on the insights gained from seismic PRAs examined in Reference 4." Reference 4 in the enclosure to the LAR is the EPRI report. The same section also states that based on those insights, "little to no anticipated unique seismic insights" are likely from the licensee's categorization process. However, the proposed Calvert Cliffs Tier 1 approach appears to exclude any qualitative and/or quantitative consideration of the mitigation capabilities of SSCs during seismically-induced events, as well as seismic failure modes that can prevent an SSC from performing its function (the technical and regulatory issues on these topics are identified in questions 1 through 5). The lack of such qualitative or quantitative determinations will place a high emphasis on the technical acceptability of the PRAs used for the case studies to develop the insights supporting the application in order to determine the acceptability and applicability of insights and to determine that technical acceptability issues with the PRAs will not affect insights such that the licensee's application would be impacted.

Section 3.3, "Demonstration of Technical Adequacy of the PRA," of RG 1.200, Revision 2, identifies two aspects necessary to demonstrate the technical acceptability of the PRA. The first aspect is assurance that the pieces of the PRA used in the application have been performed in a technically correct manner. Section 3.3.1, "Assessment that the PRA Model is Technically Correct," of RG 1.200, Revision 2, further discusses that various consensus PRA standards and industry PRA programs, as endorsed, may be interpreted to be adequate for demonstrating that the first aspect (1, above) is met.

Sections 3.3 through 3.5 of the EPRI report provide general information about the peer-reviews conducted for the PRAs used for in each of the four case studies. However, the level of information is insufficient to determine whether the pieces of the PRA used in the application have been performed in a technically correct manner.

9. For Plant A:

   a. Information available to the staff about the SPRA for Plant A includes investigation of the impact of refinement of highest acceleration (%G8) 'bin.' The results demonstrated an appreciable impact of such a refinement with a 17 percent increase in seismic LERF. As a result, it is expected that the importance measures for SSCs based on the sensitivity will be different from the base case.

Information available to the staff about the SPRA for Plant A also indicates that human error probabilities (HEPs) for FLEX actions were not considered to be failed for the highest acceleration bin. Substantial uncertainty exists about the feasibility of FLEX actions during a seismic event at acceleration levels far above the design basis. Factors such as environmental conditions, ability to clear debris, equipment status, and status of connecting locations for FLEX equipment contribute to such uncertainty.

The refinement of the highest 'bin' for seismic LERF determination, as well as the credit for FLEX actions in that bin, have the potential of impacting the dominant risk contributors and the corresponding importance measures, and therefore, the insights used to support the proposed approach.

Discuss the impact of the simultaneous refinement of highest acceleration (%G8) 'bin' and proper adjustment of HEPs associated with FLEX credit for that 'bin,' especially changes to the insights from Plant A and identification of any unique HSS SSCs from that SPRA that are not identified by the corresponding FPIE or fire PRAs.

b. The description and basis of the finding level Facts and Observation (F&O) 3-1 indicate that the approach taken at the time of the peer-review to identify dominant contributors for possible improvements was lacking realism. The suggested resolution recommends using an approach to determine potentially significant seismic failures that considers the combined impact of the sets of failures. The disposition discusses "numerous improvements" related to human reliability analysis (HRA) refinement, credit for FLEX equipment and actions, and refinement of fragility determination. However, it is unclear whether these changes were included in Plant A's SPRA used to develop the insights supporting Calvert Cliffs' proposed approach. Further, it is unclear whether a systematic approach was followed by Plant A to identify the potentially significant seismic failures that considers the combined impact of the sets of failures. The lack of a systematic approach to identify changes indicated in the F&O and/or the lack of inclusion of the changes in the SPRA during the case study has the potential of changing the categorization from the SPRA, and therefore, the insights from the case study for Plant A supporting the licensee's proposed approach.

   i. Confirm that the changes made to the SPRA to disposition F&O 3-1 were included in the Plant A SPRA used for the case study supporting the licensee's proposed approach. If the changes were not included, justify the validity and applicability of the insights from the Plant A case study, given that the changes can impact the insights and/or generate new insights.

   ii. Justify that the approach used to disposition F&O 3-1 addresses the concern of the F&O such that additional changes to the SPRA would not change the insights from the SPRA, and therefore, the case study for Plant A supporting this application.

c. Based on the information available to the staff, Plan A committed to updating its internal events PRA model to account for the requirement for two emergency diesel generator (EDG) cooling fans during periods when the outdoor temperature at the site is above the design temperature of 80 degrees Fahrenheit (°F) for the risk-informed categorization of SSCs. The staff notes that seismic events result in the likely loss of offsite power that increases the importance of EDGs, and the cooling fan success criteria results in a failure mode that can have non-negligible contribution at low seismic accelerations. Therefore:

i. Confirm that the model update of the EDG cooling fan success criteria was included in the internal events PRA, as well as the SPRA used to develop the insights from the case study. Alternately, justify that exclusion of this update from either the internal events PRA or the SPRA, or both, would not change the insights from the case study for Plant A. One way to provide this justification is to perform a sensitivity study of the updated modeling.

ii. If justification for minimal impact on insights from the case study for Plant A cannot be provided, then provide insights and their consideration in the proposed approach.

d. Based on information available to the staff, Plant A committed to removing credit for core melt arrest in-vessel at high reactor pressure vessel pressure conditions from its internal events PRA for the risk-informed categorization of SSCs.

i. Confirm that the model update of the core melt arrest in-vessel at high reactor pressure vessel pressure was included in the internal events PRA, as well as the SPRA used to develop the insights from the case study. Alternately, justify that exclusion of this update from either the internal events PRA or the SPRA, or both, would not change the insights from the case study for Plant A. One way to provide this justification is to perform a sensitivity study of the updated modeling.

ii. If justification for minimal impact on insights from the case study for Plant A cannot be provided, then provide insights and their consideration in the proposed approach.

e. Based on information available to the staff, Plant A committed to several implementation items related to its fire PRA for the risk-informed categorization of SSCs.

i. Confirm that the model updates from relevant implementation items were included in the fire PRA used to develop the insights from the case study. Alternately, justify that exclusion of this update from the fire PRA would not change the insights from the case study for Plant A. One way to provide this justification is to perform a sensitivity study of the updated modeling.

ii. If justification for minimal impact on insights from the case study for Plant A cannot be provided, then provide insights and their consideration in the proposed approach.

10. For Plant B:

a. Section 3.3.3, "Full-Power Internal Events PRA High Safety Significant Evaluation," of the EPRI report states that the importance data from the FPIE

PRA for Plant B was taken from the latest PRA input to the Maintenance Rule. Further, based on the information presented at the October 4, 2018, pre-submittal meeting by the licensee, Plant B's FPIE PRA has been used to support an integrated leak rate test frequency extension LAR. Meeting Capability Category (CC)-I of the 2009 ASME/ANS PRA Standard is considered sufficient by the NRC staff for integrated leak rate test frequency extension requests. However, consistent with RG 1.200, Revision 2, CC-II is considered acceptable for PRAs used for categorization under 10 CFR 50.69. It appears that the NRC has not reviewed the acceptability of Plan B's FPIE PRA as part of a risk-informed application for which CC-II is to be met. Furthermore, it appears that Plant B's FPIE PRA has not been subjected to an NRC accepted finding closure process. Therefore, it is unclear whether the FPIE PRA is acceptable for developing insights from the case study.

Confirm the acceptability of FPIE for deriving insights related to categorization to support the Calvert Cliffs approach by demonstrating that Plant B's FPIE PRA meets CC-II. Using NRC accepted processes, discuss how potential findings against CC-II have been resolved.

b. Section 3.3.1, "Introduction," of the EPRI report states that Plant B's SPRA includes an upgrade related to the credit for the "low leakage Flowserve N-9000 seals," and that a peer-review for the upgrade "will be scheduled in 2018." Based on information available to the staff, a finding level F&O (F&O 25-9) also states that the low leakage seal model was included in the internal events model and not peer-reviewed either. Peer-reviewers have cited the inclusion of the low leakage seal model as a PRA upgrade requiring a focused-scope peer-review according to the 2009 ASME/ANS PRA Standard. Plant B, in its resolution of the F&O, states that the finding will remain open until a focused-scope peer-review is performed. Further, information available to the staff indicates that the peer-review for the PRA upgrade has not been performed. Therefore, it is unclear whether the credit for the low leakage seals in the Plant B's FPIE PRA and SPRA has been included in accordance with acceptable regulatory processes.

   Perform a focused-scope peer-review of the low leakage seal model and report the results therefrom on the insights for identifying any unique HSS SSCs from SPRAs, or closeout any finding level F&Os using an NRC-accepted process. Alternatively, justify that the exclusion of the low leakage seal model from the Plant B's internal events and SPRA does not impact the insights derived from the case study to support the proposed approach.

c. Based on information available to the staff, a finding level F&O (F&O 25-13) for Plant B states that, "Surrogate values are used to capture the contribution for the unique nature of the actions taken in FLEX that are outside the scope of the HRA calculator." Based on this brief explanation, it is unclear how FLEX HEPs were determined. Further, the information available to the staff also includes a sensitivity study that was performed in which the FLEX HEPs were raised by a factor of 5. However, it is unclear whether the sensitivity study was performed for Plant B's internal events PRA and whether the sensitivity is sufficient to address the "surrogate approach".

i. Justify using the surrogate approach yields results that are similar to HEPs that would have considered plant-specific performance shaping factors (e.g., environmental conditions, quality of procedures, and operator training). Alternatively, justify that the sensitivity study (i.e., the factor of 5 increase) is sufficient to address the uncertainty associated with surrogate values used for HEPs associated with FLEX strategies in FPIE and seismic PRAs.

ii. Clarify whether the sensitivity study was performed for the Plant B's FPIE PRA. If not, justify that not performing the sensitivity study for the FPIE PRA will not impact the insights from the Plant B case study supporting this application and/or generate new insights. If a sensitivity study was not performed for the Plant B's FPIE PRA, and the lack of sensitivity analysis is not justified, update Plant B insights by performing simultaneous sensitivity studies on seismic and FPIE PRAs.

d. Information available to the staff about the SPRA for Plant B includes the results of sensitivity studies performed on key SPRA model assumptions and uncertainties. Among these studies is a sensitivity study on FLEX diesel generator (DG) failure rates to determine the impact on seismic risk if the failure rates are higher than they are assumed to be in the SPRA. The NRC staff notes that while industry failure rates for portable FLEX equipment are not yet available, they are expected to be different from (e.g., greater than) the failure rates for permanently installed equipment. In addition, the NRC staff notes that the failure rates for the safety-related EDGs are less than the failure rates for the station blackout DGs that are not safety-related. The information available to the NRC staff also indicates that the failure rates used in the SPRA for the FLEX DGs appear to be based on industry failure rates for EDGs. Therefore, it is unclear whether appropriate failure rates for FLEX DGs are used in the Plant B SPRA and FPIE PRA.

i. Justify the acceptability of the multiplier in the sensitivity study, either the value currently used or new proposed value, to address the uncertainty associated with using EDG failure rates for the FLEX DGs, recognizing that the FLEX DGs are neither safety-related nor permanently installed.

ii. Discuss the impact of the sensitivity study, using the multiplier justified in item i for the uncertainty associated with the failure rates for the FLEX DGs on the insights from Plant B and identification of any unique HSS SSCs from that SPRA that are not 'captured' by the corresponding FPIE PRA.

11. <u>For Plant C</u>:

a. Based on information available to the staff, it appears that the modeling of low leakage shutdown seals (SDS) is different between Plant C's FPIE and seismic PRAs. Specifically, the approach to modeling SDS behavior, and consequently, plant response under asymmetric steam generator cooling conditions appears to have been performed differently. The difference is modeling can also extend to the licensee's fire PRA. It is unclear whether modeling of SDS is consistent in Plant C PRAs and how the potential differences between PRA models may affect the insights developed from the case study using Plant C.

b. Justify that the insights developed from the case study using Plant C to support this application are not impacted by the difference in modeling the SDS behavior noted above between the licensee's internal events, fire, and seismic PRAs.

12. <u>For Plant D</u>:

a. According to the guidance in RG 1.200, Revision 2, peer-reviews against endorsed standards accounting for staff's regulatory positions on those standards, and using endorsed or accepted peer-review guidance, is an acceptable approach to demonstrate that the PRA is adequate to support a risk-informed application. Section 3.5, "Plant D Trial Categorization Evaluation," of the EPRI report provides information about the case study performed using the SPRA and FPIE PRA for Plant D. However, information regarding the peer-reviews performed and the results therefrom for those PRAs is unavailable. Therefore, the staff does not have an adequate basis to determine the technical acceptability of the PRAs used for the Plant D case study.

Provide information about peer-reviews performed for the FPIE and seismic PRAs for Plant D, as well as the status of the finding level F&Os from the corresponding peer-reviews, to support the technical acceptability of those PRAs for the case study supporting this application. Include justification that dispositions of any open F&Os do not impact the insights from the Plant D case study and/or generate new insights.

b. The discussion in Section 3.5 for Plant D states that Plant D has FLEX equipment explicitly modeled in its PRAs, including its SPRA. The NRC memorandum dated May 30, 2017, "Assessment of the Nuclear Energy Institute 16-06, 'Crediting Mitigating Strategies in Risk-Informed Decision Making,' Guidance for Risk-Informed Changes to Plants Licensing Basis" (ADAMS Accession No. ML17031A269), provides the NRC's staff assessment of challenges to incorporating FLEX equipment and strategies into a PRA model in support of risk-informed decision-making in accordance with the guidance of RG 1.200, Revision 2. The EPRI report, as well as information available to the staff, does not provide any discussion on the modeling approach, including human reliability analysis and failure probabilities for the FLEX equipment in PRAs for Plant D used to develop the insights. Therefore, the staff is unclear about the acceptability of those approaches.

i. Provide details of the methodology used to assess the failure probabilities of any modeled FLEX equipment (portable and permanently installed). Include a justification explaining the approach for estimating parameter values and consistency of the approach with the relevant supporting requirements in the 2009 ASME/ANS PRA Standard, as endorsed by RG 1.200, Revision 2.

ii. Discuss the methodology used to assess operator actions related to FLEX equipment. The discussion should include:

A. A summary of how Plant D evaluated the impact of the plant-specific human error probabilities and associated scenario-specific performance shaping factors listed in (a)-(j) of supporting requirement HR-G3 of the ASME/ANS RA-Sa-2009 PRA standard.

B. Whether maintenance procedures for the portable equipment were reviewed for possible pre-initiator human failures that render the equipment unavailable during an event, and if the probabilities of the pre-initiator human failure events were assessed as described in HLR-HR-D of the ASME/ANS RA-Sa-2009 PRA standard.

C. If Plant D's procedures governing the initiation or entry into mitigating strategies are ambiguous, vague, or not explicit, a discussion detailing the technical bases for probability of failure to initiate mitigating strategies.

13. Section 3.3 of RG 1.200, Revision 2, identifies two aspects necessary to demonstrate the technical acceptability of the PRA. The second aspect is assurance that the assumptions and approximations used in developing the PRA are appropriate. Section 3.3.2, "Assessment of Assumptions and Approximations," of RG 1.200, Revision 2, further discusses the second aspect and clarifies that, "For each application that calls upon this regulatory guide, the applicant identifies the key assumptions and approximations relevant to that application. This will be used to identify sensitivity studies as input to the decision-making associated with the application." Revision 2 of RG 1.200 defines the terms "key assumption" and "key source of uncertainty" in Section 3.3.2.

The EPRI report does not include information related to the identification of key assumptions and approximations the PRAs used in each of the four case studies and the impact of the identified key assumptions and approximations on the risk insights derived from the case studies.

a. Describe the approach used to identify and characterize the key assumptions and key sources of uncertainty in each of the PRA models used for the case studies that could affect the risk insights developed by those case studies.

b. Discuss how the potential impact of identified key assumptions and key source of uncertainty in the PRAs (i.e., internal events, seismic, and, as applicable, internal fire) on the risk insights derived from the case studies were evaluated.

c. Discuss the potential impact of the identified key assumptions and key sources of uncertainty on the risk insights and how the potential impact will be considered during implementation of the Calvert Cliffs' Tier 1 approach.

Mapping Performed to Develop Insights from Case Studies

14. 'Mapping' of HSS SSCs between SPRAs and FPIE, as well as fire PRAs, is an important aspect of the four case studies. The risk insights derived from the case studies are dependent on such 'mapping.' The mapping performed for each plant that was part of the case study is discussed in Sections 3.2 through 3.5 of the EPRI report. The following questions are related to the 'mapping' performed to arrive at the risk insights.

a. The approach for determining the importance measures for SSCs from the SPRA for seismically-induced failures is discussed for all the case study plants in Sections 3.2 through 3.5 of the EPRI report. However, there is no discussion of how the importance measures for seismically-induced and random failures were combined to generate the final importance measure for use in developing the categorization insights.

   Provide details of how the seismically-induced and random failures were combined. Include information about the consistency of the approach across the case study plants. If such a combination was not performed, justify that the insights developed from the case studies supporting this application are not impacted, and new insights are not generated for this application.

b. In several cases, passive components such as tanks are mapped to operator actions such as those involving manipulation of valves to 'align' them to the tank. Examples of such mapping include the condensate storage tank (CST) for Plant A and the main control room electrical panels for Plant B. While operator actions to manipulate valves do constitute an implicitly modeled component according to the NEI 00-04 guidance, it represents a component (i.e., valve) distinct from the passive component (e.g., tank) being mapped in the case studies. Categorization following the guidance in NEI 00-04 is performed on a component basis. Therefore, it is unclear whether the mapping discussed above was performed correctly by subsuming a HSS SSC that is uniquely identified by the SPRA.

   Justify the mapping of HSS SSCs from the SPRA to different, as well as distinct components in the FPIE and/or fire PRA, to support the insights derived from the case studies. Alternatively, update the insights derived from the case studies as identified in Section 3.6, "Summary of Sensitivity Study Insights," of the EPRI report.

c. Tables 3-6, 3-8, and 3-10 of the EPRI report contain discussions of the mapping of passive or implicitly modeled SSCs for case study Plants B through D. The discussion indicates that the seismic fragility groups that model building failures were mapped to basic events in the FPIE PRA that represent failure of the SSCs within the building, typically the common cause failure (CCF) of the SSCs. However, the mechanics of such mapping, as well as the consequences, are unclear. Further, the report (Section 3.2.5, "Comparison of Seismic PRA Results to Other PRA Results for High Safety Significant SSCs," and Table 3-4) lacks a discussion of the approach used to map building failures for Plant A. Given that buildings have multiple SSCs within them, seismically-induced building failure would impact each SSC in the buildings. It is unclear whether mapping the seismically-induced building failure event in an SPRA to one SSC that is found to be HSS (by either individual or CCF event) from the FPIE PRA would capture the impact of building failure on the remaining SSCs, especially if such SSCs are LSS.

   i. Discuss how an SSC (or SSCs) within a building under consideration was identified for mapping the seismically-induced building failure, given that buildings have multiple SSCs within them, all of which may not have CCF basic events in FPIE and some of which may be LSS.

ii. Discuss the approach used to map building failures for Plant A. Justify any differences in the approach followed by Plant A as compared to Plants B through D. The justification should include the impact on the differences, if any, on the risk insights derived from the case studies.

d. The discussion in Tables 3-6 and 3-8 of the EPRI report indicates that containment penetrations are mapped to the plant damage state in the FPIE that represents "direct LERF caused by containment bypass." Therefore, it appears that the mapping is performed to the end state and not to SSCs. It is unclear how the mapping can capture the safety significance of the impacted SSCs such as electrical and mechanical containment penetrations, fuel transfer tubes, and containment hatches. Further, it is unclear how containment penetration failures for Plants A and D were mapped.

   i. Discuss how the HSS categorization of SSCs relevant to containment penetration failures from the SPRA is captured by the mapping to the end state.

   ii. Discuss the approach used to map containment penetration failures for Plant A and justify any differences in the approach followed by Plants A and D, as compared to Plants B and C. The justification should include the impact on the differences, if any, on the risk insights derived from the case studies.

e. Section 3.5.5.3, "Seismic Fragility Groups and Common Cause Failure," of the EPRI report discusses aspects of the mapping performed for the Plant D case study. The discussion states that mapping of the seismic fragility groups to the corresponding basic events in the FPIE PRA was performed if individual failure modes for an SSC were HSS by themselves; otherwise, the fragility group was mapped to the CCF basic event. The importance measure thresholds in NEI 00-04 for individual basic events (failure modes) is different and noticeably lower than that for CCF. Therefore, mapping seismically-induced failures of individual components to CCFs of such components may prevent identification of such seismically-induced failures as unique insights.

   i. Describe and justify the process for identification and mapping of basic events representing seismically-induced failures to common cause events in the FPIE PRA. The justification should include a discussion of whether and how non-correlated failures were mapped to common cause basic events.

   ii. The discussion about mapping to common cause basic events is discussed only for Plant D and not for the other plants in the case studies. Discuss whether the approach for mapping basic events representing seismically-induced failures to common cause events in the FPIE PRA was used consistently across all the case studies. If the approach was not used for case studies besides Plant D, justify that the inconsistency in the mapping approach will not impact the insights from the case studies or generate new insights.

SUBJECT: CALVERT CLIFFS NUCLEAR POWER PLANT, UNITS 1 AND 2 – AUDIT PLAN IN SUPPORT OF REVIEW OF LICENSE AMENDMENT REQUEST TO ALLOW USE OF RISK-INFORMED CATEGORIZATION AND TREATMENT OF STRUCTURES, SYSTEMS, AND COMPONENTS (EPID L-2018-LLA-0482) DATED MARCH 20, 2019

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