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June 11, 2019

MEMORANDUM TO: File

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

SUBJECT: CALVERT CLIFFS NUCLEAR POWER PLANT, UNITS 1 AND 2 – CHANGES TO DRAFT REQUEST FOR ADDITIONAL INFORMATION REGARDING REQUEST TO ADOPT 10 CFR 50.69, "RISK-INFORMED CATEGORIZATION AND TREATMENT OF SYSTEMS, STRUCTURES AND COMPONENTS FOR NUCLEAR POWER REACTORS" (EPID L-2018-LLA-0482)

By letter dated November 28, 2018, as supplemented by letters dated November 29, 2018, and May 10, 2019 (Agencywide Documents Access and Management System (ADAMS) Accession Nos. ML18333A022, ML18337A038, and ML19130A180, respectively), Exelon Generation Company, LLC (the licensee) submitted a license amendment request (LAR) regarding 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 would 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 [Structures, Systems and Components] Categorization Guideline," dated July 2005, 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 reviewed the information provided in the LAR and determined that additional information is needed to complete its review. The enclosed Draft Request for Additional Information (RAI) was sent to the licensee by e-mail on May 17, 2019. A clarification call was held on May 23, 2019. The purpose of the clarification call was to ensure that:

- the draft RAI questions are understandable,
- the bases for the questions are clear, and
- to determine whether the information being requested was previously docketed.

In response to the licensee's comments during the clarification call, and prior to issuance, RAI 6 was changed to read:

> Section 50.69(c)(v) of 10 CFR 50 requires that the categorization be performed for entire systems and structures, not for selected components within a system or structure.

NEI 00-04, Revision 0, "10 CFR 50.69 SSC Categorization Guideline," dated July 2005 (ADAMS Accession No. ML052910035), Section 7.1, specifies sensitivity studies to be conducted for each PRA model. The sensitivity studies Section 7.1, states:

> Due to the overlap of functions and components, a significant number of components support multiple functions. In this case, the SSC or part thereof should be assigned the highest risk significance for any function that the SSC or part thereof supports.

The LAR does not discuss consideration or implementation of the guidance in NEI 00-04, Section 7.1.

NEI 00-04, Section 4, also states that a candidate low safety significance (LSS) SSC that supports an interfacing system, "will remain uncategorized until the interfacing system is considered." It further concludes, "Therefore the SSC will remain uncategorized and continue to receive its current level of treatment requirements."

NEI 00-04, Section 4, provides the following example that highlights the categorization process, which involves SSCs that support interfacing systems:

> ... cooling water piping on a ventilation system cooler is designated as part of the ventilation system. The impact of failure of the SSC on the ventilation system can be considered, but the impact of failure of the SSC on the cooling water system cannot be fully assessed until that system is considered as part of the future categorization process. Therefore, the SSC will remain uncategorized and continue to receive its current level of treatment requirements.

Consistent with 10 CFR 50.69(c)(v), which requires that the categorization be performed for entire systems and structures, the NRC staff interprets the NEI 00-04 guidance that interfacing functions/SSCs cannot be categorized and be subject to alternative treatment until the categorization of all the systems that they support is completed. Further, the SSCs supporting multiple functions will be assigned the highest risk significance for any of the functions they support.

Confirm that in the Calvert Cliffs categorization process, any functions/SSCs that serve as an interface between two or more systems will not be categorized and will not receive alternative treatment prior to completing the categorization for all of the systems that they support. Alternatively, describe and provide detailed technical and regulatory justification for your proposed approach.

The final RAIs, including revised RAI 6, were issued to the licensee by e-mail on June 4, 2019.

Docket Nos. 50-317 and 50-318

Enclosure:
Draft Request for Additional Information

cc: Listserv

DRAFT REQUEST FOR ADDITIONAL INFORMATION

REGARDING REQUEST TO ADOPT 10 CFR 50.69, "RISK-INFORMED CATEGORIZATION

AND TREATMENT OF SYSTEMS, STRUCTURES AND COMPONENTS"

EXELON GENERATION COMPANY, LLC

CALVERT CLIFFS NUCLEAR POWER PLANT, UNITS 1 AND 2

DOCKET NOS. 50-317 AND 50-318

Background

Section 50.69(b)(2)(ii) of Title 10 of the *Code of Federal Regulations* (10 CFR) 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 structures, systems, and components (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 probabilistic risk assessment (PRA).

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 (Agencywide Documents Access and Management System (ADAMS) Accession No. ML061090627), endorses, with clarifications and qualifications, Nuclear Energy Institute (NEI) 00-04, Revision 0, "10 CFR 50.69 SSC Categorization Guideline," dated July 2005 (ADAMS Accession No. ML052910035), 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 SSCs and for categorizing them into the four risk-informed safety class 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 for different approaches, depending on the scope of the 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 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," dated 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).

Draft Requests for Additional Information (RAIs)

1. Section 3.2.3 of the enclosure to the LAR states, "This approach relies on the insights gained from seismic PRAs examined in Reference 4." Reference 4 in the enclosure to the LAR is the Electric Power Research Institute (EPRI) Report 3002012988, "Alternative Approaches for Addressing Seismic Risk in 10 CFR 50.69 Risk-Informed Categorization," dated July 2018.

   The EPRI report derives risk insights from four case studies. Those case studies compare the high safety significance (HSS) SSCs determined based on a seismic PRA (SPRA) against HSS SSCs determined from other PRAs used for categorization. Each of the cases studies included a full power internal events (FPIE) PRA, but only two of the four case studies used information from a fire PRA.

   Section 3.3 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 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 is met.

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

   <u>Plant A</u>:

   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 large early release frequency (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 Diverse and Flexible Coping Strategies (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 and 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 PRA.

b. Based on information available to the staff about the SPRA for Plant A, the description and basis of the finding level Facts and Observation (F&O) 3-1 for Plant A SPRA indicates that the approach taken at the time of the peer review to identify dominant contributors for possible improvements was lacking realism. The suggested resolution for the F&O 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 refinement, credit for FLEX equipment and actions, and refinement of fragility determination. However, it is unclear whether these changes were included in the Plant A SPRA used to develop the insights supporting Exelon's 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 above-cited F&O and/or the lack of inclusion of the changes in the SPRA during the case study have 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 Plant A SPRA to disposition F&O 3-1 were included in the 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 for the Plant A SPRA 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, Plant A committed to updating its internal events PRA model for the risk-informed categorization of SSCs to (1) 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), and (2) remove credit for in-vessel core melt arrest at high reactor pressure vessel (RPV) pressure conditions. The staff notes that (1) seismic events result in the likely loss of offsite power to the cooling fan success criteria results in a failure mode for EDGs that can have non-negligible contribution at low seismic accelerations, and (2) credit for in-vessel core melt arrest at high RPV pressure conditions can impact the large early release sequences. As a result, both the updates cited above have the potential of impacting the categorization insights supporting this application from the Plant A case study.

    i. Confirm that both of the model updates cited above were included in the internal events PRA, as well as the SPRA used to develop the insights from the Plant A case study. Alternately, provide justification, such as performing a sensitivity study that simultaneously includes both of the updated modelling items cited above, that exclusion of these updates from either the FPIE or the SPRA, or both, would not change the insights from the Plant A case study.

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

Plant C:

d. Based on information available to the staff, it appears that the modeling of low leakage shutdown seals (SDS) is different between the Plant C FPIE and SPRA. 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 in modeling can also extend to the Plant C fire PRA. It is unclear whether modeling of SDS is consistent in the Plant C PRAs and how the potential differences between PRA models may affect the insights developed from the case study using Plant C.

i. Provide justification, such as performing a sensitivity study with consistent modeling of SDS across all PRAs used for the Plant C case study, that the insights developed from that case study (i.e., SSCs related to limited seismic event specific failure modes are identified that are HSS only from the SPRA and the remainder of the HSS SSCs from SPRA are captured by FPIE and/or fire PRA) are not impacted by the difference in modeling the SDS behavior noted above between the Plant C FPIE, SPRA, and fire PRA.

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

Plant D:

e. According to the guidance in RG 1.200, Revision 2, peer reviews against endorsed standards accounting for the 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 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.

i. Provide information about the status of the finding level F&Os from the peer reviews for the FPIE and SPRA to support the technical acceptability of those PRAs for the Plant D case study supporting this application.

ii. Provide justification that dispositions of any open F&Os do not impact the insights from the Plant D case study and/or generate new insights. If justification for minimal impact on insights from the Plant D case study cannot be provided, then provide updated insights and discuss their consideration in the proposed approach.

f. 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.

   i. Provide details of the methodology used to assess the failure probabilities of FLEX equipment credited in the PRAs that are dissimilar to other plant equipment credited in those PRAs (i.e., SSCs with sufficient plant-specific or generic industry data). Include a justification explaining (1) the approach for estimating parameter values, (2) the potential use of safety-related equipment failure probabilities, and (3) consistency of the approach with the relevant surveillance requirements in the 2009 ASME/ANS PRA standard, as endorsed by RG 1.200, Revision 2. One way to provide the justification for use of safety-related equipment failure probabilities is to perform a sensitivity study that increases the failure probability for modeled FLEX equipment that is dissimilar to other plant equipment credited in those PRAs used for the Plant D case study to determine the impact on the insights from the Plant D case study.

   ii. If safety-related equipment failure probabilities are used for FLEX equipment credited in the PRAs that are dissimilar to other plant equipment, and justification for minimal impact on insights from the Plant D case study cannot be provided, then provide updated insights and discuss their consideration in the proposed approach.

2. 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 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 for the PRAs used in each case study and the impact of the identified key assumptions and approximations on the insights derived from the corresponding case studies.
The NRC staff notes that information related to key assumptions and sources of uncertainty in the context of 10 CFR 50.69 categorization has been provided separately (ADAMS Accession No. ML17243A014 for Plant A, ADAMS Accession Nos. ML12248A035 and ML18052B342 for Plant C, and ADAMS Accession No. ML18334A363 for Plant D).

a. Confirm:

    i. The applicability of the key assumptions and sources of uncertainty in the above-cited documents to the corresponding case studies supporting this application.

    ii. No additional key assumptions and sources of uncertainty have been identified that are relevant to the corresponding case studies supporting this application. Provide the requested information separately for Plants A, C, and D.

b. Discuss:

    i. The potential impact on the case study insights from the key assumptions and key sources of uncertainty identified in the above-cited documents, as well as the response to item (a) of this request.

    ii. How the potential impact will be considered in the proposed alternate seismic approach. Provide the requested information separately for Plants A, C, and D.

3. 'Mapping' of HSS SSCs between SPRA and FPIE, as well as fire PRA, 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 case study is discussed in Sections 3.2 through 3.5 in the EPRI report. The following requests are related to the 'mapping' performed to arrive at the risk insights. As applicable, the requested information should be provided separately for Plants A, C, and D.

    a. The approach for determining the importance measures for SSCs from the SPRA for seismically-induced failures is discussed for case study Plants A, C, and D in Sections 3.2, 3.4, and 3.5, respectively, 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.

    For case study Plants A, C, and D:

        i. Provide details, with justification, of how the seismically-induced and random failures were combined.

ii. 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' the valves to the tank. An example of such mapping is the condensate storage tank for Plant A. While operator actions to manipulate valves does 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 an HSS SSC that is uniquely identified by the SPRA.

   For case study Plants A, C, and D, 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 of the EPRI report and discuss their consideration in the proposed approach.

c. Tables 3-8 and 3-10 of the EPRI report contain discussions of the mapping of passive or implicitly modeled SSCs for case study Plants C and 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 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. However, review of Tables 3-9 and 3-11 of the EPRI report indicates that building failures were not HSS, and therefore, did not need to be mapped to any SSCs in the FPIE.

   It is unclear whether mapping a 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 was used for case study Plants A, C, and D. Further, if such mapping was used, it is unclear whether such an approach would capture the impact of building failure on the remaining SSCs, especially if such SSCs are of LSS.

   i. Clarify whether mapping 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 was used for any or all of case study Plants A, C, and D.

   ii. If such mapping was not used for any or all of case study Plants A, C, and D, explain the intent of the discussion of such mapping in Tables 3-8 and 3-10 of the EPRI report in the context of the insights from the case studies and the proposed alternate seismic approach.

   iii. If such mapping was employed for any or all of case study Plants A, C, and D, 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.

iv. If such mapping was employed for any or all of case study Plants A, C, and D, 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 C and D. The justification should include the impact of 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, with justification, 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 Plants A and D. Justify any differences in the approach followed by Plants A and D, as compared to Plant C. The justification should include the impact of the differences, if any, on the risk insights derived from the case studies.

4. Section 3.2.3 of Attachment 1 to the letter dated May 10, 2019, includes a discussion of the consideration of seismic events for SSCs that are HSS uniquely from a PRA model but not from the integrated importance measure assessment. The discussion in the first paragraph on page 16 of Attachment 1states:

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

In the letter dated November 28, 2018, Exelon proposed to use an internal events PRA and a fire PRA for its 10 CFR 50.69 categorization at Calvert Cliffs Nuclear Power Plant, Units 1 and 2 (Calvert Cliffs). The guidance in NEI 00-04 states that HSS SSCs identified from the internal events PRA remain HSS, and the Integrated Decision-Making Panel (IDP) cannot change such SSCs to LSS. The guidance is also reflected in Table 3-1 in Attachment 1 to the supplement. Therefore, only the fire PRA model appears to be relevant to the discussion about consideration of seismic events for SSCs that are HSS uniquely from a PRA model but not from the integrated importance measure assessment.

However, the third paragraph on page 16 of Attachment 1 to the letter dated May 10, 2019, specifically identifies the consideration of seismic events for SSCs that are HSS from the fire PRA but not HSS from internal events PRA and states that for such SSCs:

> [...] the categorization team will review design-basis functions during seismic events or functions credited for mitigation and prevention of severe accidents caused by seismic events and characterize these for presentation to the IDP as additional qualitative inputs [...]

It appears that the discussion in Section 3.2.3 of Attachment 1 to the supplement provides seemingly different approaches for the consideration of seismic events for SSCs that are HSS from the fire PRA that are not HSS from the internal events PRA (i.e., SSCs that are uniquely HSS from a PRA model but not from the integrated importance measure assessment). The approach described in the first paragraph on page 16 involves "using non-PRA based qualitative assessments in conjunction with any seismic insights provided by the PRA." The other approach described in the second paragraph on page 16 involves "review [...] and characterize these for presentation to the IDP as additional qualitative inputs." It is unclear which approach is being proposed for such SSCs.

a. Clarify the approach for the consideration of seismic events for SSCs that are HSS from the fire PRA that are not HSS from the internal events PRA (i.e., SSCs that are uniquely HSS from a PRA model but not from the integrated importance measure assessment). The clarification should reconcile the seemingly different approaches noted above.

b. If the approach includes the use of "seismic insights provided by the PRA," clarify how such insights will be developed, given that Exelon does not have a seismic PRA for Calvert Cliffs.

5. NEI 00-04 provides guidance on including external events in the categorization of each SSC to be categorized. The process begins with the SSC selected for categorization, as illustrated in NEI 00-04, Section 5.4, Figure 5-6, and proceeds through the flow logic for each external hazard. According to Figure 5-6, if a component participates in a screened scenario, for that component to be considered candidate LSS, it has to be further shown that if the component was removed, the screened scenario would not become unscreened.

   LAR Section 3.2.4 indicates that all other hazards were screened from applicability to Calvert Cliffs per a plant-specific evaluation in accordance with the criteria in Part 6 of ASME/ANS PRA Standard RA-SA-2009. This statement appears to indicate that the licensee proposes to treat all SSCs as LSS with respect to other external events risk. The LAR provides no further explanation of how the risk for other external hazards will be considered in the 10 CFR 50.69 categorization (i.e., components being categorized that participate in screened scenarios and whose failure would result in an unscreened scenario). LAR Attachments 4 and 5 provide a summary of the other external hazards screening results, but do not appear to address any considerations related to applying Figure 5-6 of NEI 00-04.

   Confirm that any SSCs credited for screening of external hazards will be evaluated according to the flow chart in NEI 00-04, Figure 5-6, during the implementation of the categorization process at Calvert Cliffs, or otherwise provide technical and regulatory justification for your proposed approach.

6. NEI 00-04, Section 7.1, states, "Due to the overlap of functions and components, a significant number of components support multiple functions. In this case, the SSC, or part thereof, should be assigned the highest risk significance for any function that the SSC or part thereof supports." Section 4 of NEI 00-04 also states that a candidate LSS SSC that supports an interfacing system should remain uncategorized until all interfacing systems are categorized. The LAR does not discuss consideration or implementation of the guidance in Section 7.1 of NEI 00-04.

   Confirm that the cited guidance in NEI 00-04 will be followed and that any functions/SSCs that serve as an interface between two or more systems will not be categorized until the categorization for all of the systems that they support is completed and that SSCs that support multiple functions will be assigned the highest risk significance for any of the functions they support, or otherwise provide technical and regulatory justification for your proposed approach.

7. Paragraphs 50.69(c)(1)(i) and (ii) of 10 CFR require a licensee's PRA to be of sufficient quality and level of detail to support the SSC categorization process and that 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. The guidance in NEI 00-04 specifies sensitivity studies to be conducted for each PRA model. The sensitivity studies are performed to ensure that assumptions associated with these uncertainty parameters (e.g., human error, CCF, and failure probabilities) do not mask the SSC(s) importance.

   LAR Section 4.1 identifies RG 1.174, Revision 2, "An Approach for Using Probabilistic Risk Assessment in Risk-Informed Decisions on Plant-Specific Changes to the Licensing Basis" (ADAMS Accession No. ML100910006), dated May 2011, as an applicable regulatory requirement/criterion. RG 1.174 has been updated to Revision 3, dated January 2018 (ADAMS Accession No. ML17317A256). RG 1.174, Revision 3, cites NUREG-1855, Revision 1, "Guidance on the Treatment of Uncertainties Associated with PRAs in Risk-Informed Decision-Making, Final Report," dated March 2017 (ADAMS Accession No. ML17062A466), as related guidance. In Section B of RG 1.174, Revision 3, the guidance acknowledges specific revisions of NUREG-1855 to include changes associated with expanding the discussion of uncertainties. LAR Section 3.2.7 states that the detailed process of identifying, characterizing, and qualitative screening of model uncertainties is found in Section 5.3 of NUREG-1855, Revision 0, Volume 1, "Guidance on the Treatment of Uncertainties Associated with PRAs in Risk-Informed Decision Making, Main Report," dated March 2009 (ADAMS Accession No. ML090970525), and Section 3.1.1 of EPRI Technical Report 1016737, "Treatment of Parameter and Model Uncertainty for Probabilistic Risk Assessments." The NRC staff notes that for the FPIE (includes internal flooding) and fire PRA models, only 1 and 16 sources of uncertainty were identified, respectively.

   The NRC staff notes that NUREG-1855, Revision 1, provides guidance in Stages A through E for how to treat uncertainties associated with PRA models in risk-informed decision-making. NUREG-1855, Revision 1, cites EPRI TR 1026511, "Practical Guidance on the Use of Probabilistic Risk Assessment in Risk-Informed Applications with a Focus on the Treatment of Uncertainties."

Additionally, Section 3.3.2 of RG 1.200, Revision 2, defines key assumptions and sources of uncertainty as follows:

> A key assumption is one that is made in response to a key source of model uncertainty in the knowledge that a different reasonable alternative assumption would produce different results, or an assumption that results in an approximation made for modeling convenience in the knowledge that a more detailed model would produce different results. For the base PRA, the term "different results" refers to a change in the risk profile (e.g., total CDF and total LERF, the set of initiating events and accident sequences that contribute most to CDF and to LERF) and the associated changes in insights derived from the changes in the risk profile. A "reasonable alternative" assumption is one that has broad acceptance within the technical community and for which the technical basis for consideration is at least as sound as that of the assumption being challenged.

A key source of uncertainty is one that is related to an issue in which there is no consensus approach or model and where the choice of approach or model is known to have an impact on the risk profile (e.g., total CDF and total LERF, the set of initiating events and accident sequences that contribute most to CDF and to LERF) such that it influences a decision being made using the PRA. Such an impact might occur, for example, by introducing a new functional accident sequence or a change to the overall CDF or LERF estimates significant enough to affect insights gained from the PRA.

Based on the information provided in the LAR, the NRC staff requests the following information to confirm the key assumptions and sources of uncertainty provided for the 10 CFR 50.69 risk-informed application in Attachment 6 of the LAR were properly assessed from the base PRAs that have received peer reviews. The NRC staff requests that the licensee provide the following:

a. A brief description of the process and the criteria used to identify, from the initial comprehensive list of uncertainties and assumptions for the base PRA model(s) (including those associated with plant specific features, modeling choices, and generic industry concerns), the application-specific key assumptions and sources of uncertainties provided in LAR Attachment 6. Describe how the key assumptions and sources of uncertainty are determined consistent with the definitions in RG 1.200, Revision 2.

b. Provide a summary list of any new key assumptions and sources of uncertainty that have been identified for the application as a result of resolving question 9.a and discuss how each newly identified key assumption and uncertainty will be dispositioned in the categorization process. The discussion should clarify whether the licensee is following the guidance in Section 5 of NEI 00-04 by performing sensitivity analysis or other accepted guidance such as NUREG-1855.

c. Confirm that the process described in question 9.a is consistent with NUREG-1855, Revision 1, or other NRC-accepted methods (e.g., NUREG-1855, Revision 0). If deviating from the current guidance provided in NUREG-1855, Revision 1, provide a basis to justify the methods use in the 10 CFR 50.69 categorization process (e.g., exclusion/consideration of EPRI TR 1026511).

8. The NRC memorandum dated May 30, 2017, provides the NRC's staff assessment of identified differences between NEI 16-06 guidance and the guidance in RG 1.200, Revision 2, for incorporating diverse and flexible (FLEX) coping strategies and equipment into a PRA model in support of risk-informed decision-making. It is unclear whether the licensee's FPIE and/or fire PRA used to support this application credit FLEX equipment and operator actions. For the NRC staff to determine the acceptability of incorporation of FLEX equipment into the PRA model(s), provide the following:

    a. Confirm whether FLEX equipment and associated operator actions have been credited in the FPIE and/or fire PRA. If no FLEX credit is applied in the FPIE and/or fire PRA, the remaining questions for this RAI do not apply.

    b. If FLEX equipment or operator actions have been credited in the PRA, address the following separately for FPIE (includes internal flooding), fire PRA, and external hazards screening, as appropriate:

        i. Summarize the supplemental equipment and compensatory actions, including FLEX strategies that have been quantitatively credited for each of the PRA models used to support this application. Include discussion of whether the credited FLEX equipment is portable or permanently installed equipment.

        ii. Discuss whether the credited equipment (regardless of whether it is portable or permanently installed) is similar to other plant equipment (i.e., SSCs with sufficient plant-specific or generic industry data) and whether credited operators actions are similar to other operator actions evaluated using approaches consistent with the endorsed ASME/ANS RA-Sa-2009 PRA standard.

        iii. If any credited FLEX equipment is dissimilar to other plant equipment credited in the PRA (i.e., SSCs with sufficient plant-specific or generic industry data), discuss the data and failure probabilities used to support the modeling and provide the rationale for using the chosen data. Discuss whether the uncertainties associated with the parameter values are in accordance with the ASME/ANS PRA standard, as endorsed by RG 1.200, Revision 2.

        iv. If any operator actions related to FLEX equipment are evaluated using approaches that are not consistent with the endorsed ASME/ANS RA-Sa-2009 PRA standard (e.g., using surrogates), discuss the methodology used to assess human error probabilities for these operator actions. The discussion should include:

            1. A summary of how 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 was evaluated.

2. 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.

3. If the 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.

v. The ASME/ANS RA-Sa-2009 PRA standard defines PRA upgrade as the incorporation into a PRA model of a new methodology or significant changes in scope or capability that impact the significant accident sequences or the significant accident progression sequences. Section 1-5 of Part 1 of the ASME/ANS RA-Sa-2009 PRA standard states that upgrades of a PRA shall receive a peer review in accordance with the requirements specified in the peer review section of each respective part of this standard.

   Provide an evaluation of the model changes associated with incorporating mitigating strategies that demonstrates none of the following criteria is satisfied: (1) use of new methodology, (2) change in scope that impacts the significant accident sequences or the significant accident progression sequences, and (3) change in capability that impacts the significant accident sequences or the significant accident progression sequences.

SUBJECT: CALVERT CLIFFS NUCLEAR POWER PLANT, UNITS 1 AND 2 – CHANGES TO DRAFT REQUEST FOR ADDITIONAL INFORMATION REGARDING REQUEST TO ADOPT 10 CFR 50.69, "RISK-INFORMED CATEGORIZATION AND TREATMENT OF SYSTEMS, STRUCTURES AND COMPONENTS FOR NUCLEAR POWER REACTORS" (EPID L-2018-LLA-0482) DATED JUNE 11, 2019

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