NRR-DMPSPEm Resource

From: Wiebe, Joel
Sent: Wednesday, May 9, 2018 7:39 AM
To: Ryan Sprengel
Subject: Preliminary RAIs for Braidwood and Byron Stations 50.69 Amendment

Ryan,

Preliminary RAIs are provided to ensure a common understanding of the requests. If you need a teleconference to clarify any of the RAIs, let me know by May 16, 2018. A response to the RAIs is requested within 30 days.

Joel

Title 10, of the Code of Federal Regulations, Part 50, Section 69 (10 CFR 50.69), "Risk-Informed Categorization and Treatment of Structures, Systems, and Components for Nuclear Power Reactors", allows licensees to use a risk-informed process to categorize systems, structures, and components (SSCs) according to their safety significance in order to remove SSCs of low safety significance from the scope of certain identified special treatment requirements. Regulatory Guide (RG) 1.201, Revision 1, "Guidelines for Categorizing Structures, Systems, and Components in Nuclear Power Plants According to their Safety Significance" (Agencywide Documents Access and Management System (ADAMS) Accession No. ML061090627) endorses, with regulatory positions and clarifications, the Nuclear Energy Institute (NEI) guidance document NEI 00-04, Revision 0 "10 CFR 50.69 SSC Categorization Guideline", (ADAMS accession No. ML052910035) as one acceptable method for use in complying with the requirements in 10 CFR 50.69. Both RG 1.201 and NEI 00-04 cite RG 1.200, "An Approach for Determining the Technical Adequacy of Probabilistic Risk Assessment Results for Risk-Informed Activities," (ADAMS Accession No. ML040630078) which endorses industry consensus probabilistic risk assessment (PRA) standards, as the basis against which peer reviews evaluate the technical adequacy of a PRA. Revision 2 of RG 1.200 is available at ADAMS Accession No. ML090410014.

By letter dated September 1, 2017 (ADAMS Accession No. ML17244A093), as supplemented by letter dated April 4, 2018 (ADAMS Accession No. ML18094A955), Exelon Generation Company, LLC (Exelon), submitted a license amendment request (LAR) to adopt 10 CFR 50.69, Risk-informed Categorization and Treatment of Structures, Systems, and Components for Nuclear Power Reactors at Byron Station Units 1 and 2 (Byron), and at Braidwood Station, Units 1 and 2 (Braidwood). Section 3.1.1 of the LAR states that Exelon will implement the risk categorization process in accordance with NEI 00-04, Revision 0, as endorsed by RG 1.201. However, the licensee's LAR does not contain enough information for the Nuclear Regulatory Commission (NRC) staff to determine if the licensee has implemented the guidance in NEI 00-04, as endorsed by RG 1.201, appropriately as a means to demonstrate compliance with all of the requirements in 10 CFR 50.69. The following requests for additional information (RAIs) outline the specific issues and information needed to complete the NRC staff's review:

RAI 1 Scope and Quality of PRA Self-Assessments and Peer Reviews

10 CFR 50.69(c)(i) requires that a licensee's PRA must be of sufficient quality and level of detail to support the categorization process, and must be subjected to a peer review process assessed against a standard or set of acceptance criteria that is endorsed by the NRC. Section 3.3 of the LAR states that the internal events probabilistic risk assessments (IEPRAs) for Byron and Braidwood were "subject to a self-assessment and a full-scope peer review" in July 2013, and that the fire PRAs (FPRAs) were "subject to a self-assessment and full-scope peer review" for Byron in June 2015 and for Braidwood in October 2015. The scope and reason for the self-assessments are not described in the LAR. Also, Attachment 3 of the LAR did not identify which plant to attribute the findings and observations (F&Os). Address the following:

a) Describe the scope and reason for the self-assessment performed for the IEPRAs and FPRAs. If the self-assessment(s) were needed to meet the guidance in RG 1.200, Revision 2, provide detailed information on the self-assessment(s), including any resulting F&Os, if not already provided in the LAR.

b) Confirm that the IEPRA full-scope peer review included internal flooding.

c) The findings presented in Attachment 3 of the LAR included self-assessment and full-scope peer review findings. Identify which F&Os are a result of the self-assessment and which were a result of the full-scope peer review.

d) Identify the site that each F&O corresponds to and which F&Os pertain to both sites.

RAI 2 Facts and Observations (F&O) Closure Process

10 CFR 50.69(b)(2)(iii) requires that the results of the peer review process conducted to meet 10 CFR 50.69 (c)(1)(i) criteria be submitted as part of the LAR. Section 3.3 of the LAR indicates that in February 2017 an F&O closure process was performed for both the Byron and Braidwood IEPRA, internal flood PRA (IFPRA), and FPRA F&Os, and as a result several findings were closed. The February 2017 "closure review" was performed prior to the NRC's acceptance of this guidance on May 3, 2017. Provide the following information to confirm that the F&O closure review was performed consistent with the Appendix X (ADAMS Package Accession No. ML17086A431) to NEI 05-04, 07-12, 12-13 guidance concerning the process for "Close Out of Facts and Observations" that the staff accepted, with conditions, in the letter dated May 3, 2017 (ADAMS Accession Number ML17079A427).

a) Clarify whether a focused-scope peer review was performed concurrently with the F&O closure process. If so, provide the following:

   i. Summary of the scope of the peer review.

   ii. Detailed descriptions of any new findings generated from the peer review and their disposition for the application.

b) Confirm that the licensee provided the closure review team a written assessment and justification of whether the resolution of each F&O, within the scope of the independent assessment, constitutes a PRA upgrade or maintenance update, as defined in American Society of Mechanical Engineers/American Nuclear Society (ASME/ANS) RA-Sa-2009, "Standard for Level 1/Large Early Release Frequency Probabilistic Risk Assessment for Nuclear Power Plant Applications, Addendum A to RA-S-2008," as endorsed by RG 1.200 Revision 2.

c) Section X.1.3 of the Appendix X guidance includes the following five criteria for selecting members of the closure review team:

   i. Every member of the independent assessment team should be independent of the PRA associated with the F&Os being reviewed, per the criteria of "independent" in the ASME/ANS PRA Standard. These members may be contractors, utility personnel, or employees of other utilities, and may include members of peer review teams that previously reviewed the models being assessed.

   ii. Every member of the independent assessment group should meet the relevant peer reviewer qualifications as stated in the ASME/ANS PRA Standard for the technical elements associated with the F&Os being reviewed.

   iii. The overall review team experience includes two qualified reviewers for each F&O. An exception to this is allowed for the closure of an F&O related to a single SR, in which case, a single independent reviewer is acceptable, in alignment with the peer review guidance in the main body of this document and in accordance with the ASME/ANS PRA Standard.

iv. Each member of the independent assessment team should be knowledgeable about the F&O independent assessment process used to assess the adequacy of the F&O resolution.

v. The total number of reviewers is a function of the scope and number of finding F&Os to be reviewed for closure

Describe how the selection of members for the February 2017 independent assessment met this criteria.

d) Explain how closure of the F&Os was assessed to ensure that the capabilities of the PRA elements, or portions of the PRA within the elements, associated with the closed F&Os now meet capability category-II (CC-II) for all the applicable supporting requirements (SRs) of ASME/ANS RA-Sa-2009 as endorsed by RG 1.200 Revision 2.

e) Discuss whether the F&O closure review scope included all finding-level F&Os, including those finding-level F&Os that are associated with "Met" SRs. If not, identify and provide detailed descriptions for any F&Os that were excluded from the F&O closure review scope, and their disposition for the application.

<u>RAI 3 Open/Partially Open Findings in the Process of Being Resolved</u>

10 CFR 50.69(c)(i) requires that a licensee's PRA must be of sufficient quality and level of detail to support the categorization process, and must be subjected to a peer review process assessed against a standard or set of acceptance criteria that is endorsed by the NRC. Section 50.69(b)(2)(iii) of 10 CFR requires that the results of the peer review process conducted to meet 10 CFR 50.69 (c)(1)(i) criteria be submitted as part of the application. During a peer review, the documentation of differences or deficiencies between the licensee's PRA model and the NRC endorsed PRA standard are labeled as F&Os. Attachment 3 of the LAR, "Disposition and Resolution of Open Peer Review Finding and Self-assessment Open Items," provides F&Os and self-assessment findings that are still open or only partially resolved after the February 2017 F&O closure review. Address the following:

a) <u>F&O SY-B12-01</u> regarding High Energy Line Break (HELB) scenarios

The disposition states that there is heating ventilation and air conditioning (HVAC) dependency in HELB scenarios that is not modeled in the PRAs. It further states that a sensitivity study shows this exclusion to have only a small impact on core damage frequency (CDF) and large early release frequency (LERF), and that this increase would "not trigger consideration of an emergent model update per Exelon Risk Management procedures." NRC staff notes that though this modeling exclusion may have a small impact on the total risk, its inclusion could potentially increase the risk importance values for certain system components above the NEI 00-04, Section 5 threshold criteria for determining high safety significance (HSS).

i. Justify that the HVAC dependency modeling exclusion cited above does not impact the results of the 10 CFR 50.69 categorization process.

ii. Alternatively, propose a mechanism to ensure that this HVAC dependency modeling is incorporated into the PRA prior to implementation of the 10 CFR 50.69 risk categorization process.

b) <u>F&O 16-4</u> regarding breaker coordination calculation results

The disposition states that the results of the breaker coordination calculations not available at the time of the F&O closure “will be incorporated into the current FPRA update.” The disposition does not appear to provide analysis or identify any additional requirements as discussed in the original F&O.

i. Describe the results of the assessment performed to confirm that breaker coordination is adequate for circuits credited in the FPRA. Include confirmation that the assessment was performed in accordance with guidance in NUREG/CR-6850, “EPRI/NRC RES Fire PRA Methodology for Nuclear Power Facilities,” (ADAMS Accession Nos. ML052580075, ML052580118, and ML103090242) including credit for cable length.

ii. Identify the circuits that could not be confirmed to be coordinated and explain how these inadequacies are resolved for this application.

iii. If updated PRA modeling was used to resolve this inadequacy, then describe and justify the updated modelling of the uncoordinated circuits. Include description of the component failure modes assumed upstream and downstream of possible fault locations in the uncoordinated circuit.

iv. If the disposition will not be complete at the issuing of the RAI response, propose a mechanism to ensure that the breaker coordination modeling is incorporated into the PRA prior to implementation of the 10 CFR 50.69 risk categorization process.

c) <u>F&O 20-8 (SR FSS-B2)</u> regarding Main Control Room (MCR) abandonment modeling

The disposition indicates that the approach to crediting alternate shutdown given abandonment of the MCR relies on scaling the sequence Conditional Core Damage Probability (CCDP) / Conditional Large Early Release Probability (CLERP) based on the complexity of the shutdown rather than on actual modelling of fire-induced damage, hardware failures, and operator errors. As such, this approach cannot inform the determination of importance measures for components whose failures are not modelled. The disposition states that this approach will be updated and reviewed after the guidance in NUREG-1921, Supplement 1, “EPRI/NRC-RES Fire Human Reliability Analysis Guidelines: Qualitative Guidance for Main Control Room Abandonment Scenarios,” (ADAMS Accession No. ML16110A413), for treating MCR abandonment scenarios is issued. The disposition also states that resolution of this issue will have no impact on the application given the “limited contribution” of MCR abandonment to total fire risk. The NRC staff notes that modelling issues that can result in even a small impact to CDF and LERF, can potentially increase the risk importance values for certain system components above the NEI 00-04 Section 5 threshold criteria for determining safety significance. In light of these observations:

i. Explain how MCR abandonment scenarios were modelled using “scaling factors,” and justify that the treatment is adequate for this application. To augment this explanation, include:

- An explanation of how the “scaling factors” were estimated;
- An explanation of how the failure of SSCs that could be categorized are modelled in MCR abandonment scenarios using this approach;
- An explanation of how MCR abandonment scenarios due to loss of control versus loss of habitability were modelled. Include discussion of how the decision to abandon the MCR due to loss of control was addressed;
- A description of which operator actions were assumed to be required and how these action were determined; and,

- An explanation of how the times available for needed operator actions and the times required to perform the actions were determined.

ii. Alternatively, justify why the treatment of MCR abandonment scenarios using “scaling factors” rather than fault tree modeling will have no impact on the application. Include discussion of the FPRA CDF/LERF contribution of these scenarios and the impact that the modelling can have on the categorization of components that participate in these scenarios.

d) F&O 26-9 (SR IGN-A7) regarding improperly screened wall-mounted cabinets

The disposition states that the risk increase due to incorporating wall-mounted panels with greater than four switches into the FPRA is expected to be offset by the risk decrease due to the reduction in fire ignition frequency caused by the higher panel count in fire scenarios of higher risk. The NRC staff notes that this risk trade-off is not valid if the risk associated with the wall mounted panels is significant.

i. Justify that the risk increase due to incorporating wall-mounted panels with greater than four switches into the FPRAs is offset by the risk decrease due to the reduction in fire ignition frequency caused by the higher panel count. Include discussion of the component failures that would be caused by fire-induced damage of the excluded wall mounted panels and their risk significance.

ii. If the justification in item i. above cannot be provided, then provide other justification or propose a mechanism that ensures that the excluded wall-mounted panels are incorporated into the FPRAs prior to implementation of the 10 CFR 50.69 categorization process.

e) F&O 25-11 regarding SR PRM-B2 disposition of IE F&O in the FPRA

The F&O closure review states in the disposition to this F&O that the treatment of sump [screen] clogging has been updated for the IEPRA consistent with the guidance provided in WCAP-16362-NP, “PRA Modeling Template for Sump. Blockage,” Revision 0, April 2005, to resolve an F&O. However, this treatment has not yet been updated in the FPRA. The disposition states that this failure mode is considered to have a minimal impact on the FPRA and therefore will not be updated until the next revision of the FPRA.

i. Justify that treatment of sump clogging in the FPRA has a minimal impact on the 10 CFR 50.69 application. Include an explanation of how the sump clogging is currently modeled.

ii. If the justification in item i. above cannot be provided, then propose a mechanism that ensures that the treatment of sump clogging is updated in the FPRAs prior to implementation of the 10 CFR 50.69 categorization process.

f) F&O 25-5 regarding significant contributor review

The disposition for F&O 25-5 does not match the finding. The finding is related to review of significant fire risk contributors, but the disposition discusses refinement of joint error probability values and treatment of containment isolation valves for the mini-purge lines. This appears to be a typographical error. Please reconcile this apparent mismatch between F&O 25-5 and the disposition provided in LAR Attachment 3.

RAI 4 PRA maintenance versus PRA upgrade

10 CFR 50.69(b)(2)(iii) requires that the results of the peer review process conducted to meet 10 CFR 50.69 (c)(1)(i) criteria be submitted as part of the LAR. During a peer review, the documentation of differences or deficiencies between the licensee’s PRA model and the NRC endorsed PRA standard are labeled as facts and observations (F&Os). Section 3.3 of the LAR states “[a]ll the models described below have been peer reviewed

and there are no PRA upgrades that have not been peer reviewed.” Confirm that any PRA update performed to resolve an F&O discussed in RAI 3 or an uncertainty issue discussed in RAI 8, does not constitute a “PRA upgrade” as defined in ASME/ANS RA-Sa-2009.

RAI 5 Overall Categorization Process

10 CFR 50.69(b)(2)(i) requires that a licensee’s application contain a description of the process for SSC categorization. LAR Section 3.1.1 “Overall Categorization Process,” has two different sets of bulleted elements and concludes with an additional list of ten elements. Some of the elements discuss training that will be given, some discuss the different hazard models, and some discuss PRA model results. It is not clear from these discussions what the sequence of evaluations will be in the categorization process, what information will be developed and used, and what guidance on acceptable decisions by the Integrated Decision-making Panel (IDP) will be followed during the categorization of each system. Information on the training and expertise of the IDP team is provided in the LAR and need not be repeated in the response to this RAI.

a) Summarize, in the order they will be performed, the sequence of elements or steps that will be followed for each system that will be categorized. A flow chart, such as that provided in the NEI presentation (ADAMS Accession No. ML17249A072) for the September 6, 2017, public meeting with NEI regarding 10 CFR 50.69 LARs (ADAMS Accession No. ML17265A020) may be provided instead of a description. The steps should include:
   i. The input from all PRA evaluations such as use of the results from the IEPRA, IFPRA, and FPRA;
   ii. The input from non-PRA approaches (seismic, other external events, and shutdown);
   iii. The input from the responses to the seven qualitative questions in Section 9.2 of NEI 00-04;
   iv. The input from the defense-in-depth (DID) matrix;
   v. The input from the passive categorization methodology.
b) In the response to item a) above, clarify the difference between “preliminary HSS” and “assigned HSS” and identify which inputs can, and which cannot, be changed from preliminary HSS to low safety significance (LSS) by the IDP, and confirm that the proposed approach is consistent with the guidance in NEI 00-04 as endorsed by RG 1.201.
c) In the response to item a) above, clarify which steps of the process are performed at the function level and which steps are performed at the component level. Describe how the categorization of the component impacts the categorization of the function, and vice-versa. Describe instances in which the final safety significance of the function would differ from the safety significance of the component(s) that support the function, and confirm that the proposed approach is consistent with the guidance in NEI 00-04 as endorsed by RG 1.201.
d) Section 7 of NEI 00-04 states that “if any SSC is safety significant, from either the PRA-based component safety significance assessment (Section 5) or the DID assessment (Section 6), then the associated system function is preliminary safety significant.” The NRC staff interprets that the cited guidance applies to all aspects identified in Sections 5 and 6 of NEI 00-04, including Section 5.3 through 5.5 dedicated to seismic, external hazards, or shutdown risk.

If the licensee’s categorization process differs from the guidance in Section 7 of NEI 00-04 cited above where functions supported by any HSS component(s) will be assigned HSS, describe and justify the approach.

e) The industry flow chart presented at the September 6, 2017 public meeting shows that the passive categorization would be undertaken separately from the active categorization.
   i. Explain how the results from the passive categorization will be integrated with the overall categorization results.
   ii. If the results from the passive categorization can be changed by the IDP, explain and justify the proposed approach.

RAI 6 SSCs Categorization based on Other External Hazards

10 CFR 50.69(c)(1)(i) and (ii) require that a licensee's SSC categorization process consider results and insights from a plant-specific PRA that is of sufficient quality and level of detail to support the SSC categorization process, as well as 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 guidance in NEI 00-04 provides one acceptable method for including external events in the categorization of each SSC to be categorized. Fire and Seismic hazards are discussed in Section 5.2 and 5.3 respectively. All other hazards are discussed in Section 5.4 "Assessment of Other External Hazards". Figure 5-6 in Section 5.4 illustrates the process that begins with the SSC selected for categorization and then proceeds through the flow chart for each external hazard. Figure 5-6 shows that if a component participates in a screened scenario, then in order 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 states that the, "The Braidwood Station and Byron Station categorization process will use screening results from the Individual Plant Evaluation of External Events (IPEEE) in response to GL 88-20, "Individual Plant Examination for Severe Accident Vulnerabilities," for evaluation of safety significance related to the [following] other external hazards." LAR Section 3.2.4 continues that ""[a]ll SSCs credited in other IPEEE external hazards are considered HSS." The use of "other" instead of a more precise description does not allow the NRC staff to compare the license's proposed process with the guidance.

a) Identify the external hazards that will be evaluated according to the flow chart in Figure 5-6 of NEI 00-04.
b) Identify which hazards will have ""[a]ll SSCs credited […] considered HSS" instead of using the flow chart.
c) Describe and justify any additional method(s) different from (a) or (b) above, that will be used to evaluate individual SSCs against external hazards and identify the hazards that will be evaluated with these methods.
d) Confirm that all hazards not included in the categorization process (a), (b), or (c) above, will be considered insignificant for every SSC and therefore will not be considered during the categorization process.
e) Attachment 4 of the LAR indicates that extreme wind or tornado hazards are screened. With regards to extreme wind or tornado hazards, address the following:
    i. Identify what type of SSCs, if any, are credited in the screening of these hazards, such as passive or active features.

    ii. If any SSCs are credited for screening, then explain and justify how the guidance in Figure 5-6 of NEI 00-04 will apply to external flooding hazards.

RAI 7 Shutdown Risk

10 CFR 50.69(c)(1)(ii) requires that a licensee's SSC categorization process 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. LAR Section 3.2.5 states the Braidwood Station and Byron Station categorization process will use the shutdown safety management plan described in NUMARC 91-06, "Guidelines for Industry Actions to Assess Shutdown Management," (ADAMS Accession No. ML14365A203), for categorization of safety significance related to low power shutdown conditions. However, the LAR does not cite the other criteria specified in NEI 00-04 Section 5.5 pertaining to low power shutdown events (i.e., includes DID attributes and failures that would initiate a shutdown event). Clarify and provide a basis for how the categorization of SSCs will be performed for shutdown events, and how it is consistent with the guidance in NEI 00-04 as endorsed by RG 1.201.

RAI 8 Key Assumptions and Uncertainties that could Impact the Application

10 CFR 50.69(c)(1)(i) and (ii) require that a licensee's PRA 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 dispositions presented in Attachment 6 of the LAR for key assumptions and modeling uncertainties state in each case that: "this does not represent a key source of uncertainty and will not be an issue for the 50.69 calculations." However, in a number of instances there is not enough information provided in the dispositions for the NRC staff to determine whether the uncertainty will not impact 10 CFR 50.69 risk categorization. In light of these observations address the following:

a) <u>Diesel Generator Cooling Fan Success Criteria</u>

Attachment 6 (page 54) of the LAR states that, at times, the outdoor air temperature at the plant would require two diesel generator (DG) cooling fans for the DGs. However, the LAR states that just one-of-two DG cooling fans is assumed to be an adequate success criterion for all times of the year in the PRA, because plant procedures provide guidance for emergency restoration of DG ventilation to maintain acceptable temperatures for the DGs. The disposition states that this emergency ventilation is conservatively not credited in the PRA. Based on publically available information, the high temperatures at each of the plant locations have reached highs above 90° Fahrenheit (F) between the months of April and October. Braidwood have reached highs over 95° F between the months of May and September, and over 100° F between June and August. Byron has reached highs over 100° F between May and September. It appears that the assumed one-of-two DG fan success criteria may be challenged a significant portion the year. In light of these observations:

i. Provide a basis for the adequacy of the assumed one-out-of-two DG cooling fans success criteria for the time periods cited above. Include an explanation of what constitutes the term 'restoration,' such as setting up temporary ventilation after a DG cooling fan fails or starting the second cooling fan or repairing a failed DG cooling fan. Provide confirmation that all operator actions associated with the restoration are proceduralized.

ii. Alternatively, quantitatively justify that the application (e.g., risk measures for DG related components) is not sensitive to the assumption that one-of-two DG cooling fans provide adequate cooling when the temperature at the plant requires two cooling fans.

b) <u>Condensate Storage Tank (CST) Refill</u>

Attachment 6 (page 59) of the LAR states that the CST inventory is insufficient for the 24 hour mission time of the PRA model. The two options available to use the auxiliary feedwater after 16 hours when the CST is depleted depend on (1) operator actions to the refill the CST, or (2) automatic transfer of the auxiliary feedwater suction to the service water system. The disposition states that neither of these two actions are included in the PRA model but their exclusion "should not have a significant impact on CDF or LERF." It appears to the NRC staff that auxiliary feedwater was assumed to be successful after 16 hours, though failures of actions that must be performed manually or automatically are not modelled. It is not clear to the NRC staff that exclusion of the cited modeling would have a minimal impact on the 10 CFR 50.69 application.

ASME/ANS-Ra-Sa-2009 states that a component may be excluded from the system model if the total failure probability of component failure modes resulting in the same effect on system operation is at least two orders of magnitude lower than the highest failure probability of the other components in the same system. In light of these observations:

i. Justify the exclusion of the cited failure modes associated with actions required to use auxiliary feedwater after 16 hours when the CST is depleted. If the screening methodology is quantitatively based, provide the quantitative criteria. If another process was used to screen these mitigation actions, then explain how the approach meets ASME/ANS RA-Sa-2009. Include in the justification why the excluded modelling cannot impact the 10 CFR 50.60 process.

If exclusion of the cited failure modes cannot be justified, then propose a mechanism that ensures that the excluded failure modes are incorporated into the PRAs before they are used in the 10 CFR 50.69 risk categorization process.

ii. Alternately, justify that the excluded modelling cannot impact the 10 CFR 50.69 process including categorization of components whose failures have been excluded.

c) Post-Fire Human Reliability Analysis

Attachment 6 (page 65) of the LAR states that the joint human error probability (HEP) dependency analysis for the FPRA is a source of uncertainty. The disposition states that minimum floor value of 1E-06 was applied for all HEP combinations. NUREG-1921 discusses the need to consider a minimum value for the joint probability of multiple human failure events (HFEs) in human reliability analyses (HRA). NUREG-1921 refers to Table 2-1 of NUREG-1792, "Good Practices for Implementing Human Reliability Analysis (HRA)," April 2005 (ADAMS Accession No. ML051160213), which recommends that joint HEP values should not be below 1 E-5. Table 4-4 of EPRI 1021081, "Establishing Minimum Acceptable Values for Probabilities of Human Failure Events," provides a lower limiting value of 1 E-6 for sequences with a very low level of dependence. Therefore, the guidance in NUREG-1921 allows for assigning joint HEPs that are less than 1 E-5, but only through assigning proper levels of dependency. In light of these observations:

i. Confirm that each joint HEP value used in the FPRA below 1E-5 includes its own justification that demonstrates the inapplicability of the NUREG-1792 lower value guideline (i.e., that the criteria for independent HFEs are met).

ii. Provide an estimate of the number of these joint HEP values below 1E-5, discuss the range of values, and provide at least two different examples where this justification is applied.

RAI 9 Passive Component Categorization Process

LAR Section 3.1.2 states that passive components and the passive function of active components will be evaluated using the method for risk-informed repair/replacement activities consistent with the safety evaluation issued by the Office of Nuclear Reactor Regulation, “Request for Alternative ANO2-R&R-004, Revision 1, Request to Use Risk-informed Safety Classification and Treatment for Repair/Replacement Activities in Class 2 and 3 Moderate and High Energy Systems, Third and Fourth 10-Year Inservice Inspection Intervals,” for Arkansas Nuclear One, Unit 2, dated April 22, 2009 (ADAMS Accession No. ML090930246). The LAR further states that this methodology will be applied to determine the safety significance of Class 1 SSCs.

The NRC staff notes that this methodology has been approved for Class 2 and Class 3 SSCs. Because Class 1 SSCs constitute principal fission product barriers as part of the reactor coolant system or containment, the consequence of pressure boundary failure for Class 1 SSCs may be different than for Class 2 and Class 3, and therefore the criteria in the ANO-2 methodology cannot automatically be generalized to Class 1 SSCs without further justification.

The LAR does not justify how the ANO-2 methodology can be applied to Class 1 SSCs and how sufficient DID and safety margins are maintained. A technical justification for Class 1 SSCs should address how the methodology is sufficiently robust to assess the safety significance of Class 1 SSCs, including, but not limited to: justification of the appropriateness of the CCDP numerical criteria used to assign ‘High’, ‘Medium’ and ‘Low’ safety significance to these loss of coolant initiating events; identification and justification of the adequacy of the additional qualitative considerations to assign ‘Medium’ safety significance (based on the CCDP) to ‘High’ safety significance; justification for crediting operator actions for success and failure of pressure boundary; guidelines and justification for selecting the appropriate break size (e.g. double ended guillotine break or smaller break); and include supporting examples of types of Class 1 SSCs that would be assigned low safety significance, etc.

As mentioned in the meeting summary from the February 20, 2018 Risk-Informed Steering Committee (RISC) meeting (ADAMS Accession No. ML18072A301), the NRC staff understands that the industry is planning to limit the scope to Class 2 and Class 3 SSCs, consistent with the pilot Vogtle license amendment (ADAMS Accession Number ML14237A034).

Please provide the requested technical justification or confirm the intent to apply the ANO-2 passive categorization methodology only to Class 2 and Class 3 equipment.

RAI 10 Unit Differences for Categorization

10 CFR 50.69(c)(i) requires that a licensee's PRA must be of sufficient quality and level of detail to support the categorization process, and must be subjected to a peer review process assessed against a standard or set of acceptance criteria that is endorsed by the NRC. The LAR includes results for the full power IEPRA, including IFPRA, as well as the FPRA for both the Byron and Braidwood sites. It appears that the 2013 peer review addressed the full power IEPRA for both sites. In Attachment 2 of the LAR, the baseline risk values indicate that the sites and units are not completely symmetrical and therefore, warrant discussion of the differences between the sites and units. Furthermore, it is not clear how a single peer review team could review the models for multiple sites unless the sites were identical.

a) Describe how the IEPRA, IFPRA, and FPRA models are maintained for the two sites, including:

   i. Discussion of how the differences between the sites and the units within the sites are modelled using one master model for the internal, internal flooding and fire events PRAs. Include a brief description of the major differences between the sites and units. If some differences are not modelled then justify not modeling these differences.

   ii. Discussion of how the peer reviews for each unit PRA were conducted in light of the fact that one master model was used.

b) Describe how categorization will be handled at each site (e.g. are sites so similar that categorization of an SSC would also apply to the other site?).

RAI 11 Modeling of the RCP Shutdown Seals

In letter dated December 2015, "Report of Full Compliance with March 12, 2012 Commission Order Modifying Licenses with Regard to Requirements for Mitigation Strategies for Beyond-Design-Basis External Events (Order Number EA-12-049)" (ADAMS Accession No. ML15350A414) and letter dated February 26, 2016 to the NRC, "Sixth Six-Month Status Report in Response to March 12, 2012 Commission Order Modifying Licenses with Regard to Requirements for Mitigation Strategies for Beyond Design-Basis External Events (Order Number EA-12-049)" (ADAMS Accession No. ML16057A209), it is stated that Braidwood Station and Byron Station have installed the Westinghouse reactor coolant pump (RCP) SHIELD Passive Thermal Shutdown Seals (SDS) (Generation III).

The PRA model for the Generation III Seals was approved by the NRC in the August 23, 2017, Topical Report [TR] PWROG-14001-P, Revision 1, 'PRA Model for the Generation III Westinghouse Shutdown Seal"' and the associated NRC Safety Evaluation (ADAMS Package Accession No. ML17200A116).

Consistent with the RG 1.174 guidance that the PRA scope, level of detail and technical acceptability be based on the as-built and as-operated and maintained plant, and reflect operating experience at the plant, please address the following:

a) Clarify whether the current internal events and fire PRA models include credit for the Westinghouse Generation III RCP seals.

b) If the PRA models include credit for the Westinghouse Generation III RCP seals, address the following:

i. Confirm that the limitations and conditions in the NRC safety evaluation for PWROG-14001-P, Revision 1, are met.

ii. If exceptions to the limitations and conditions exist, identify all the exceptions and justify impact on the application.

iii. Clarify whether the Generation III Westinghouse RCP seal model has been peer-reviewed as part of the internal events PRA and fire PRA peer-reviews.

iv. If this RCP seal model has not been peer reviewed, justify why the addition of this model is not considered a PRA upgrade requiring a focused-scope peer review.

v. If the addition of RCP seal model qualifies as a PRA upgrade, provide the results from the focused-scope peer review including the associated F&Os and their resolutions.

c) If the PRA models do not include credit for the Westinghouse Generation III RCP seals, provide justification for not modeling the seals for the application.

d) Alternatively to item c) above, propose a mechanism to implement the RCP seals model in the PRA models and ensure compliance with the TR PWROG-14001-P, Revision 1, the NRC Safety Evaluation, and associated limitations and conditions, prior to implementation of the 10 CFR 50.69 categorization process.

RAI 12 Implementation Items

10 CFR 50.69(b)(2)(ii) requires that a licensee's application contain a description of the measures taken to assure that the quality and level of detail of the systematic processes that evaluate the plant for internal and external events during normal operation, low power, and shutdown are adequate for the categorization of SSCs. If the responses to RAIs 1-11 above require any follow-up actions prior to implementation of the 10 CFR 50.69 categorization process, provide a list of those actions and any PRA modeling changes including any items that will not be completed prior to issuing the amendment but must be completed prior to implementing the 10 CFR 50.69 categorization process. Propose a mechanism that ensures these activities and changes will be completed and appropriately reviewed and any issues resolved prior to implementing the 10 CFR 50.69 categorization process (for example, a license condition that includes all applicable implementation items and a statement that they will be completed prior to implementation of the 10 CFR 50.69 categorization process).

Hearing Identifier: NRR_DMPS
Email Number: 353

Mail Envelope Properties (Joel.Wiebe@nrc.gov20180509073900)

Subject: Preliminary RAIs for Braidwood and Byron Stations 50.69 Amendment
Sent Date: 5/9/2018 7:39:27 AM
Received Date: 5/9/2018 7:39:00 AM
From: Wiebe, Joel

Created By: Joel.Wiebe@nrc.gov

Recipients:
"Ryan Sprengel" <ryan.sprengel@exeloncorp.com>
Tracking Status: None

Post Office:

Files	**Size**	**Date & Time**
MESSAGE	40250	5/9/2018 7:39:00 AM

Options
Priority: Standard
Return Notification: No
Reply Requested: No
Sensitivity: Normal
Expiration Date:
Recipients Received: