Exelon Generation®

4300 Winfield Road
Warrenville, IL 60555

630 657 2000 Office

RS-18-072

10 CFR 50.90
10 CFR 50.69

June 13, 2018

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

Braidwood Station, Units 1 and 2
Renewed Facility Operating License Nos. NPF-72 and NPF-77
NRC Docket Nos. 50-456 and 50-457

Byron Station, Units 1 and 2
Renewed Facility Operating License Nos. NPF-37 and NPF-66
NRC Docket Nos. 50-454 and 50-455

Subject: Response to Request for Additional Information Regarding Braidwood and Byron Stations 50.69 Amendment

References:

1. Letter from David M. Gullott (Exelon Generation Company, LLC) to U.S. NRC, "Application to Adopt 10 CFR 50.69, 'Risk-informed categorization and treatment of structures, systems, and components for nuclear power reactors'," dated September 1, 2017

2. Letter from David M. Gullott (Exelon Generation Company, LLC) to U.S. NRC, "Correction to License Amendment Request to Adopt 10 CFR 50.69, 'Risk-informed categorization and treatment of structures, systems, and components for nuclear power reactors'," dated April 4, 2018

3. Email from J. Wiebe (U.S. NRC) to R. Sprengel (Exelon Generation Company, LLC), "Preliminary RAIs for Braidwood and Byron Stations 50.69 Amendment," dated May 9, 2018

In Reference 1, Exelon Generation Company, LLC (EGC) requested an amendment to Renewed Facility Operating License Nos. NPF-72 and NPF-77 for Braidwood Station, Units 1 and 2, and Renewed Facility Operating License Nos. NPF-37 and NPF-66 for Byron Station, Units 1 and 2. Reference 2 provided revised text to the Reference 1 submittal. In Reference 3, the NRC requested that EGC provide additional information to support their review of the subject License Amendment Request. As noted in Reference 3, a response was requested within 30 days; i.e., by June 8, 2018. An extension was requested on June 5, 2018, to delay the response until June 13, 2018. The requested information is provided in Attachment 1.

EGC has reviewed the information supporting the No Significant Hazards Consideration and the Environmental Consideration that was previously provided to the NRC in Reference 1. The

additional information provided in this submittal does not affect the conclusion that the proposed license amendment does not involve a significant hazards consideration. This additional information also does not affect the conclusion that neither an environmental impact statement nor an environmental assessment need be prepared in support of the proposed amendment.

In accordance with 10 CFR 50.91, "Notice for public comment; State consultation," paragraph (b), EGC is notifying the State of Illinois of this additional information by transmitting a copy of this letter and its attachment to the designated State Official.
There are no regulatory commitments contained in this letter. Should you have any questions concerning this letter, please contact Ryan M. Sprengel at (630) 657-2814.

I declare under penalty of perjury that the foregoing is true and correct. Executed on the 13th day of June 2018.

Respectfully,

David M. Gullott
Manager – Licensing
Exelon Generation Company, LLC

Attachment 1: Response to Request for Additional Information

cc: Regional Administrator – NRC Region III
NRC Senior Resident Inspector – Braidwood Station
NRC Senior Resident Inspector – Byron Station
Illinois Emergency Management Agency

ATTACHMENT 1

Response to Request for Additional Information

In Reference 1, Exelon Generation Company, LLC (EGC) requested an amendment to Renewed Facility Operating License Nos. NPF-72 and NPF-77 for Braidwood Station, Units 1 and 2, and Renewed Facility Operating License Nos. NPF-37 and NPF-66 for Byron Station, Units 1 and 2. Reference 2 provided revised text to the Reference 1 submittal. In Reference 3, the NRC requested that EGC provide additional information to support their review of the subject License Amendment Request. As noted in Reference 3, a response was requested within 30 days; i.e., by June 8, 2018. An extension was requested on June 5, 2018, to delay the response until June 13, 2018. The requested information is provided below.

Request for Additional Information

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.*

Response: The 'self-assessments' were performed prior to the peer reviews in preparation for the peer reviews and were not gap assessments. These self-assessments were not needed to meet any particular purpose in RG 1.200 Revision 2. The peer reviews were full-scope, performed in accordance with RG 1.200 Revision 2 and represent the full set of F&Os for the Byron and Braidwood PRAs.

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

Response: Yes, the July 2013 IEPRA full-scope peer review included internal flooding. Also the F&O closure review in February 2017 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.*

Response: All items in Attachment 3 are from the full-scope peer reviews. See the response to sub-question (a) above for explanation of the 'self-assessments'.

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

Response: For the IEPRA (including internal flooding), all F&Os apply to both sites and units. For the Fire PRA, all F&Os were evaluated for their applicability to both sites and units, and their resolution was applied to both sites and units, as applicable.

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).

ATTACHMENT 1

Response to Request for Additional Information

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.*

Response: No focused scope peer reviews were required or performed as documented in Section 1.2 of the Byron and Braidwood F&O Closure Technical Report, "PRA FINDING LEVEL FACT AND OBSERVATION TECHNICAL REVIEW OF BYRON AND BRAIDWOOD NUCLEAR POWER PLANTS UNITS 1 AND 2, 032299-RPT-05" (Reference 4).

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.*

Response: Section 2.1.1 of the Byron and Braidwood F&O Closure Technical Report (Reference 4), documents the scope and preparation for the F&O Closure review, and states the following:

> "Expectations regarding preparation for the review (NEI 05-04, Section 4.2) and conduct of the self-assessment by the host utility (NEI 05-04, Section 4.3) were addressed prior to conduct of this review. This included documentation by the host utility of resolution of the prior PRA peer review finding-level F&Os and preparation of the information required for this independent assessment. The documented bases for F&O closure provided by the model development team included a written assessment whether the resolution constituted PRA maintenance or PRA upgrade."

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.*

Response: Section 2.1.1, Appendix B.1, and Table B-1 of the Byron and Braidwood F&O Closure Technical Report (Reference 4) acknowledge the independence requirement (Appendix X) and documents the independence of the reviewers.

ATTACHMENT 1

Response to Request for Additional Information

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.*

Response: The Byron and Braidwood F&O Closure review team consisted of a multi-disciplinary team. Section 2.1.1, Appendix B.1, and the resumes in Appendix B of the Byron and Braidwood F&O Closure Technical Report (Reference 4) acknowledge the qualifications requirement and documents the qualification of the reviewers. The relevant peer reviewer qualifications as stated in the ASME/ANS PRA Standard for the technical elements associated with the F&Os being reviewed were satisfied.

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.*

Response: Section 2.1.1, Table 3-2, and Table 3-3 of the Byron and Braidwood F&O Closure Technical Report, documents the assignment of at least two qualified reviewers to each F&O.

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.*

Response: As documented in Appendix B of the Byron and Braidwood F&O Closure Technical Report (Reference 4), most of the independent assessment team had conducted other Peer Reviews as well as F&O Closure Reviews following the Appendix X process. Additionally, the F&O Closure Review process documented in Section 2 of the Byron and Braidwood F&O Closure Technical Report was reviewed with the entire team at the start of the review.

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.

Response: Section 2 and Section 3 of the Byron and Braidwood F&O Closure Technical Report (Reference 4), documents the selection of members and shows the team consisted of eight reviewers.

The 171 F&Os were divided into ten review units, each of which was assigned to at least two of the reviewers. In general, the review units were based on technical elements, but in some cases the technical element was broken up across review units based on the specific content of the F&Os and where they fit best.

Section 3.3 and Table 3-3 of the Byron and Braidwood F&O Closure Technical Report, documents the schedule for the F&O Closure Review, and shows the distribution of the F&Os to the eight reviewers.

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.*

Response: Section 2 (Details of the Review Processes) of the Byron and Braidwood F&O Closure Technical Report (Reference 4) describes the approach taken during the review.

- The process guidance in NEI 05-04, Section 4.6, was applicable to this review.
- The independent technical review team reviewed the documented bases for closure of the finding-level F&Os prepared by the host utility.
- The independent technical review team determined whether the finding-level F&Os in question had been adequately addressed and could be closed out by consensus.
- As part of this process each F&O was reviewed regarding whether the closure response represented PRA maintenance or a PRA upgrade.
- Details of the F&O Closure review assessment are documented in Tables A-1 and A-2 of the Byron and Braidwood F&O Closure Technical Report.
- Appendix C of the Byron and Braidwood F&O Closure Technical Report provides clarification that the completion of the F&O Closure Review resulted in all closed Findings meeting 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.
- Section 2.1.4 of the F&O closure report specifically states that the closure review team concluded that all SRs where the F&Os have been closed are now MET at CC II.

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.*

Response: Section 1.1 of the Byron and Braidwood F&O Closure Technical Report (Reference 4) documents that the scope of the review consisted of "Finding" level F&Os. This included F&O's that were associated with "Met" supporting requirements.

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

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) *F&O SY-B12-01 regarding High Energy Line Break (HELB) scenarios*

ATTACHMENT 1

Response to Request for Additional Information

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.

Response: EGC will update the internal events and fire PRA models to versions that include this HVAC dependency modeling prior to implementation of the 10 CFR 50.69 risk categorization process. This change does not incorporate new methods and is not expected to result in significant changes in the risk results.

b) F&O 16-4 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.

Response: The breaker coordination review for both Byron and Braidwood was performed by reviewing the breaker coordination calculations. Where breaker coordination was confirmed, no model changes are required. The review of breaker coordination calculations resulted in identification of specific breakers and specific buses for which breaker coordination could not be demonstrated. For these buses/breakers the load cables will be modeled as additional cables causing failure of the bus. The review was performed in accordance with the requirements of NUREG/CR-6850. No credit for cable length was required since the more conservative approach of failing the bus for the uncoordinated load cables was applied. The results of this review will be incorporated into a revised Fire PRA model prior to implementation of the 10 CFR 50.69 risk categorization process. This change is only model maintenance as it applies methods already implemented in the Fire PRA and evaluated by the peer review.

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

Response: Several breakers associated with the 480 V load centers and all breakers on the 120 V AC instrument buses were found to lack adequate coordination. The associated load cables will be included in the model as cables causing failure of the bus as noted in the responses to part i of this question.

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.

Response: The updated modeling for the uncoordinated breakers consisted of listing the associated load cables against the bus as cables causing failure of the bus. This includes the total length of the load cable so that any fault anywhere along the length of the cable will result in failure of the associated bus.

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.

Response: This finding will be resolved prior to implementation of 10 CFR 50.69 for Byron and Braidwood, as noted in the responses to parts i, ii, and iii of this question.

c) F&O 20-8 (SR FSS-B2) 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:

ATTACHMENT 1

Response to Request for Additional Information

- *An explanation of how the "scaling factors" were estimated;*

Response: As described in the 50.69 LAR Attachment 3, for F&O 20-8, the methodology used was previously applied and accepted by NRC via safety evaluations for several NFPA 805 applications.

The following criteria were used for defining the scaling factor to be used for adjusting the CCDP and CLERP values for the control room abandonment scenarios:

CCDP Range	Adjusted CCDP
< 0.001	0.1
≥ 0.001 and < 0.1	0.2
>0.1	1.0

CLERP values are adjusted as follows:

CLERP adjusted = CCDP adjusted x (CLERP calculated / CCDP calculated

- *An explanation of how the failure of SSCs that could be categorized are modelled in MCR abandonment scenarios using this approach;*

Response: The CCDP and CLERP adjustment factors (CCDP adjusted / CCDP calculated and CLERP adjusted / CLERP calculated) are applied to the control room abandonment cutsets as a multiplier to all cutsets for each abandonment scenario. The importance of all basic events (BEs) in the cutset file which merges all fire scenario cutsets is increased based on the application of the scaling factor to the control room abandonment cutsets. The use of the adjusted cutset file incorporates the impact of the CCDP and CLERP scaling on the BE importances used to evaluate the SSCs to be categorized.

- *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;*

Response: Control room abandonment is evaluated for loss of habitability only. Regarding fire-induced loss of control, command and control is expected to remain in the control room and the HEPs for any credited operator actions are adjusted to account for fire.

- *A description of which operator actions were assumed to be required and how these action were determined; and,*

Response: The operator actions credited are fire adjusted HEPs credited for other, non-main control room abandonment scenarios. This includes all operator actions in the model with the exception of actions taken in the control room which are not credited since the control room is abandoned. The application of the CCDP and CLERP adjustments specified above address the impact of the transfer of command and control from the control room to the remote shutdown panel.

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

Response: The HEP time available and the time required for operator action are not altered by control room abandonment. A short delay in initiation of these actions due to control room abandonment is assumed to be accounted for by the CCDP and CLERP adjustment factors.

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.*

Response: EGC intends to retain the scaling factors at this time. To ensure that the impact of the CCDP and CLERP scaling factor adjustments is accounted for in the categorization process, a Fire PRA sensitivity in addition to the sensitivities required by NEI 00-04 Table 5-3 will be performed.

This sensitivity will consider the baseline fire CDF and LERF (without application of scaling factors to CCDPs and CLERPs) to represent lower bound values and the fire CDF and LERF with the scaling factors to represent upper bound values. The Fire PRA model input to the categorization process will include the importance measures from both sets of quantifications for consideration by the IDP. This ensures that any CCDP and CLERP scaling factor impacts are considered in the categorization. If the Fire PRA is updated in the future to eliminate the scaling factor adjustment, this sensitivity calculation would no longer be performed.

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.*

Response: Identification of all wall mounted panel configurations with four or more switches will be performed and any model changes required will be incorporated into the Fire PRA prior to implementation of the 10 CFR 50.69 risk categorization process. This change does not incorporate new methods as the approach to modeling panels is the same as in the current model which has been peer reviewed.

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.*

Response: The new sump clogging value is included in the current FPRA model update which will be issued prior to implementation of the 10 CFR 50.69 risk 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.

Response: The disposition identifies the issues that were driving the areas with significant risk contribution. Resolution of these items reduced the risk and eliminated the undeveloped scenarios from the high risk contributors. This change has been incorporated into the FPRA model that will be used for the 10 CFR 50.69 risk categorization process. This change to the Fire PRA risk model incorporated plant data to eliminate conservatism in the analysis and does not incorporate new methods not already included in the model that was peer reviewed.

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.

ATTACHMENT 1

Response to Request for Additional Information

Response: The PRA updates identified in RAI 3 and RAI 8 to resolve remaining F&Os or model uncertainties do not constitute "PRA Upgrades", for the reasons noted in those responses. Changes to address these open F&Os involve updates to some portions of the modeling using previously-reviewed methods.

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*

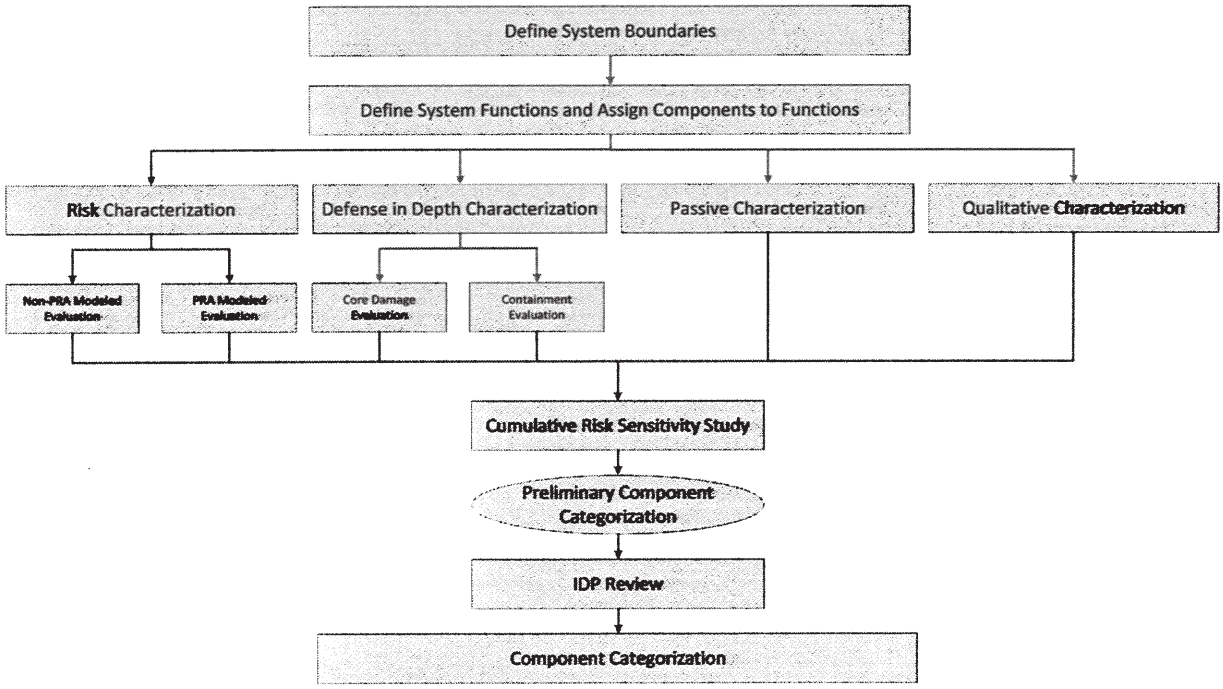
confirm that the proposed approach is consistent with the guidance in NEI 00-04 as endorsed by RG 1.201.

Response: The process to categorize each system will be consistent with the guidance in NEI 00-04, "10 CFR 50.69 SSC Categorization Guideline," as endorsed by RG 1.201. RG 1.201 states that "the implementation of all processes described in NEI 00-04 (i.e., Sections 2 through 12) is integral to providing reasonable confidence" and that "all aspects of NEI 00-04 must be followed to achieve reasonable confidence in the evaluations required by §50.69(c)(1)(iv)." However, neither RG 1.201 nor NEI 00-04 prescribe a particular sequence or order for each of the elements to be completed. Therefore, the order in which each of the elements of the categorization process (listed below) is completed is flexible and as long as they are all completed they may even be performed in parallel. Note that NEI 00-04 only requires Item 3 to be completed for components/functions categorized as LSS by all other elements. Similarly, NEI 00-04 only requires Item 4 to be completed for safety related active components/functions categorized as LSS by all other elements.

1. PRA-based evaluations (e.g., the internal events, internal flooding, and fire PRAs)
2. Non-PRA approaches (e.g., seismic safe shutdown equipment list (SSEL), other external events screening, and shutdown assessment)
3. Seven qualitative criteria in Section 9.2 of NEI 00-04
4. The defense-in-depth assessment
5. The passive categorization methodology

Below is an example of the major steps of the categorization process described in NEI 00-04:

Figure 5-1: Categorization Process Overview



ATTACHMENT 1

Response to Request for Additional Information

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

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

Table 1: Categorization Evaluation Summary

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

Passive	Passive – Section 4	Segment/ Component	No	Not Allowed

Notes:

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

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

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

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

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.

Response: Section 5 defines categorization process considerations for both PRA-based and non-PRA-based (i.e., deterministic) assessment methods. Section 5.3, for example, describes the process for categorization from seismic risk considerations using either a seismic PRA (i.e., PRA-based) or using a seismic margin assessment (SMA, i.e., deterministic and not PRA-based). Section 7 requires assigning the safety significance of functions to be preliminary HSS if it is supported by an SSC determined to be HSS from the PRA-based assessment in Section 5, but does not require this for SSCs determined to be HSS from non-PRA-based, deterministic assessments in Section 5. The interpretation of this requirement is further clarified in the Vogtle SER (ML14237A034) which states "...if any SSC is identified as HSS from either the integrated PRA component safety significance assessment (Section 5 of NEI 00-04) or the defense-in-depth assessment (Section 6), the associated system function(s) would be identified as HSS."

The reason for this is that the application of non-PRA-based assessments results in the default safety significance categorization of any SSCs associated with the safe shutdown success paths defined in those deterministic assessments to be HSS regardless of its risk significance. Therefore, there is no risk basis for assigning the SSC-associated functions to be HSS, since the deterministic analyses from which the associated safe shutdown equipment lists are derived do not define functions equivalent to those used in the categorization process. This is the reason that the guidance in Section 7 of NEI 00-04 clearly notes "PRA-based" in reference to Section 5 of NEI 00-04. The categorization process is consistent with the guidance in NEI 00-04 as endorsed by RG 1.201.

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.*

Response: Please see the response to Questions 05a, 05b, and 05c. If the results of the passive categorization are HSS, then the SSC is categorized as preliminary HSS regardless of the other categorization elements. A HSS determination by the passive categorization process cannot be changed by the IDP, as noted in the response to these RAIs.

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.*

Response: The "other" external hazards that will be evaluated according to the flow chart in Figure 5-6 of NEI 00-04 are any hazards listed in Attachment 4 of the LAR, "External Hazards Screening," that have not been screened in accordance with ASME/ANS PRA Standard RA-Sa-2009.

b) *Identify which hazards will have ""[a]ll SSCs credited [...] considered HSS" instead of using the flow chart.*

Response: The statement "All SSCs credited in other IPEEE external hazards are considered HSS" was intended to be consistent with the flow chart in Figure 5-6 of NEI 00-04. There are no Other External Hazards that will be evaluated using a method other than depicted in 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.*

Response: There are no additional method(s) different from (a) or (b) that will be used to evaluate individual SSCs against external hazards.

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.*

Response: All external hazards not included in the categorization process (a), (b), or (c) above are 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.

Response: For the Braidwood Station, there were no SSCs credited in the screening of extreme winds and tornados, including passive and/or active components, other than Seismic Category I structures.

In addition to the Seismic Category I structures for the Byron Station, the eight (8) redundant service water cooling towers (SX CTs) and their supporting SSCs (e.g., fans, riser valves, electrical switchgear) are credited in the screening of the tornado missile hazard and thus will be considered high safety significant.

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.

Response: Please see response to part i of this question.

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.

Response: For plants without a shutdown PRA, such as Braidwood and Byron, NEI 00-04, as endorsed by RG 1.201, allows the use of a process based on the NUMARC 91-06 program. Braidwood and Byron's categorization process follows the guidance and criteria in Section 5.5 in NEI 00-04 to address shutdown risk. Below is a summary of the NEI 00-04 process and requirements.

The overall process for addressing shutdown risk is illustrated in Figure 5-7 of NEI 00-04. NUMARC 91-06 specifies that a defense-in-depth approach should be used with respect to each defined shutdown key safety function. The shutdown key safety functions defined in

NUMARC 91-06 are evaluated for categorization of SSCs. NEI 00-04 provides two criteria for SSCs to be considered preliminary HSS.

1. If a system/train being categorized supports a shutdown key safety function as the primary or first alternate means, then it is considered to be a "primary shutdown safety system" and is categorized as preliminary HSS. The station's Shutdown Safety Management Program, which is consistent with NUMARC 91-06, is used as a guide to identify primary and first alternative means. NEI 00-04 defines a "primary shutdown safety system" as also having the following attributes:

    - It has a technical basis for its ability to perform the function.
    - It has margin to fulfill the safety function.
    - It does not require extensive manual manipulation to fulfill its safety function.

2. If a failure of the SSC being categorized would initiate an event during shutdown plant conditions (e.g., loss of shutdown cooling, drain down), then that SSC is categorized as preliminary HSS.

As stated in NEI 00-04, "If the component does not participate in either of these manners, then it is considered a candidate as low safety significance with respect to shutdown safety."

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) Diesel Generator Cooling Fan Success Criteria

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.

Response: There is one main ventilation fan for each individual EDG. This is the fan that is modeled as required in the PRA. Additional exhaust fans also exist, but it was originally unclear whether or not these were relied upon during hot weather. Recent interviews with the system engineer provide information that the exhaust fans are only present to prevent buildup of fumes and are not relied upon for ventilation for the EDGs. The design calculation assumes an outside dry-bulb temperature of 95°F. Though a higher outside temperature may exist briefly, the conservatisms built into the design calculation and the PRA support the current assumption. Variations around that temperature are expected, but do not invalidate the design. Such conservatisms in the design calculation include:

- The outside air temperature is assumed to be a constant 95°F with the thermal inertia of the structure already reflecting that outside temperature. On a real high-temperature day, some significant amount of time would be required before the structure also heats up to the assumed starting temperature.
- The mean daily temperature range is 21°F. A short-duration peak temperature above 95°F would not be expected to last more than a few hours, further reducing the fraction of time where the 95°F would be exceeded.
- Peak temperatures are calculated as a maximum of 125.4°F /131.4°F (EDG 1A/1B) using a theoretical heat load, and only 111.2°F/110.3 (EDG 1A/1B) with a measured heat load, with a design limit of 132°F. Temperatures above 132°F would not be expected to cause an immediate failure.

These supporting design calculations for EDG cooling find that the design basis of the plant requires that the one main ventilation fan is required, as is modeled in the PRA. The combination of conservatisms in the design analysis as described above provide the basis for the current modeling of the one fan as appropriate for a reasonably realistic PRA model.

Regarding 'restoration' of adequate cooling on a loss of EDG ventilation, several alarm response procedures (e.g., BAR/BWAR-0-31/34-A1/A2 (Reference 5)) at both sites direct operators to open the roll-up door and provide portable fans to circulate air. As discussed above, there would be substantial time available to respond to a need for increasing cooling of the EDG locations with the main ventilation fan working. Therefore the impact to the PRA model of including this response action was judged to be negligible.

Without the main ventilation fan working, the EDG is failed in the PRA.

b) *Condensate Storage Tank (CST) Refill*

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.*

Response: The 16-hour assumption for auxiliary feedwater requirements were based on a simple calculation of maximum feedwater flow for the full duration of the time frame. More recent assessments that are now in the PRA documentation show that the CST provides sufficient volume to maintain auxiliary feedwater for greater than 24 hours using realistic time-dependent feedwater flowrates. Therefore, this assumption is no longer required and will be removed from future documentation.

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).

Response: The Fire PRA to be used for the 50.69 categorization will continue to use a 1E-6 HEP floor value for the joint minimum HEP associated with dependent human failure events. For each joint human error probability (JHEP) below 1E-5, justification for the JHEP will be included in the Fire PRA dependency analysis documentation.

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.

Response: The current Fire PRA applies a minimum JHEP of 1E-6. There are on the order of 100 HFE combinations with JHEPs ranging from 1E-5 to 1E-6 in the Fire PRA for each site. In accordance with the HRA dependency process, examples of criteria where operator action can be considered to be independent are as follows:

1. the actions are separated by a successful action integral to the definition of the accident sequence.
2. the actions do not share a common cognitive (e.g. share the same cue), and are not expected to occur at the same time and will not be performed in the same location.
3. the actions are sufficiently separate in time such that a shift change is expected to occur.

The following provides two examples from the Byron FPRA which contain independent HFEs and for which the JHEP is calculated to be below 1E-5, and these are representative of Braidwood as well.

Example 1 - Combination 361. The two actions shown in combination 361 are independent because they are separated by successful action. There is a successful diagnosis of loss of Auxiliary Feedwater followed by the successful decision to implement feed and bleed between the two actions. These actions are justified as independent due to the intervening success action. Thus, this joint HEP combination would remain at 6.3E-6 which is slightly below 1E-5.

ATTACHMENT 1

Response to Request for Additional Information

BE ID	Description	Notes on the Individual Actions	Independent HEP	Modeled JHEP
1AF01PB-FO-HXVOA-F	FIRE - OPERATORS FAIL TO REFILL DDAFP FUEL OIL DAY TANK FROM STORAGE TANK	1AF01PB-FO-HXVOA-F applies to LOSP and transient scenarios where AFW successfully starts and runs. In order to keep AFW running the diesel driven AF pump day tank will need to be refilled by connecting the day tank to the diesel storage tank. The alarm for the low fuel oil is expected to occur at 3 hours and then operators have an additional 3 hours after the tank alarm to refill the tank. Upon receiving the instructions to refill the tank, the execution is estimated to take 7 minutes.	5.7E-3	
Plant response after failure to refill DDAFP FO Day Tank		Failure to refill the DDAFP FO Day Tank leads to a loss of Auxiliary Feedwater, which is successfully diagnosed and the operators successfully implement Feed and Bleed cooling.		
1RH-SP-T---HPMOA-F	FIRE - OPERATORS FAIL TO STOP RH PUMPS (TRANSIENT)	1RH-SP-T---HPMOA-F applies to transient sequences given loss of heat sink. For this scenario, the loss of heat sink occurs after failure to refill the DDAFP Day Tank. The operators have successfully recognized the loss of AF, transitioned to FR-H.1 and have successfully diagnosed the need for feed and bleed. There is a caution in FR-H.1 that notifies the operators that the RH pumps cannot be operated longer than 2.4 hours without CC flow to the RH heat exchangers. The RH pumps are assumed to start when feed and bleed cooling is initiated as the RH pumps automatically start on an SI signal, which is generated when the operator actuates SI for feed and bleed per FR-H.1 direction. Given successful bleed and feed, the RWST will be depleted over time and there will be a need to switch to ECCS recirculation. Since the RH System is not in operation during normal power operations, CC flow to the RH heat exchangers is normally secured (i.e., motor-operated valves 1/2CC9412A and 1/2CC9412B on the heat exchanger outlets are normally closed). The RH pumps are then secured within 2.4 hours if ECCS recirculation (and therefore CC to the RH HXs) is not established in that interval.	1.1E-3	6.30E-6

ATTACHMENT 1

Response to Request for Additional Information

Example 2 - Combination 166. The two actions shown in combination 166 are independent because they have separate independent cues, they are separated in time (by over an hour), and they will not be performed in the same location. The PRA conservatively uses 54 minutes to the cue for the switchover to recirculation based on the LLOCA timing calculation. However, the RCP Seal LOCA is a small LOCA so the time to switchover to recirculation would be longer and thus the best estimate timing is more than 60 minutes. These actions are justified as independent due to separate cues, separated in time, and separated locations. Thus, this JHEP combination would remain at 1.1E-6 which is below 1E-5.

BE ID	Description	Notes on the Individual Actions	Independent HEP	Modeled JHEP
1RC-PUMPS--HPMOA-F	FAILURE TO START STANDBY CCW/CCP/SX PUMPS TO MAINTAIN RCP SEAL COOLING	1RC-PUMPS--HPMOA-F models the manual start of a standby pump should the running pump trip. The fire PRA assumes that the running pump is failed by the fire at T=0 (start of the fire concurrent with reactor trip). Starting the standby pump is performed in the MCR and expected to take 1 minute to execute. Failure of this action combines with hardware failures to fail Unit 1 CCW and will eventually lead to an RCP seal LOCA failure.	1.4E-3	1.1E-6
Plant response after failure to start a standby pump to maintain RCP seal cooling.		Failure of RCP seal cooling leads to an RCP seal LOCA followed by successful high pressure injection. In addition, Unit 1 has been successfully aligned to the Unit 0 CC pump following loss of the Unit 1 CCW. The PRA assumes that the Unit 0 is always aligned to the non-impacted unit. Aligning to Unit 0 CCW will maintain Unit 1 cooling and allow for success of high pressure injection. After successful injection (and/or cooldown and depressurization) the operators would next implement the switchover to recirculation cooling on low RWST level.		
0CC-SXHTX0-HHXOA-F	FIRE - OPERATORS FAIL TO ALIGN SX TO CC HX 0 (BYRON)	The cue for switch over to recirculation is low level in the RWST. In accordance with BBW MAAP run BB0021 the criteria for recirculation for a LLOCA will be met at 54 minutes. For the RCP seal LOCA the recirculation criteria are expected to be met in greater than 1 hour. This is a local action occurring outside of the MCR, to align the train of CC cooling that was not affected by the initiating event, and it is estimated to take 10 minutes to execute.	7.9E-4	

ATTACHMENT 1

Response to Request for Additional Information

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.

Response: The passive categorization process is intended to apply the same risk-informed process accepted by the NRC in the ANO2-R&R-004 for the passive categorization of Class 2, 3, and non-Code class components. This is the same passive SSC scope the NRC has conditionally endorsed in ASME Code Cases N-660 and N-662 as published in Regulatory Guide 1.147, Revision 15. Both code cases employ a similar risk-informed safety classification of SSCs in order to change the repair/replacement requirements of the affected LSS components. All ASME Code Class 1 SSCs with a pressure retaining function, as well as supports, will be designated high safety-significant, HSS, for passive categorization which will result in HSS for its risk-informed safety classification, and cannot be changed by the IDP.

Therefore, this methodology and scope for passive categorization is acceptable and appropriate for use at Braidwood and Byron for 10 CFR 50.69 SSC categorization.

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.*

Response: The current IEPRA model is a combined PRA model that represents all the units at both sites (i.e., Byron Unit 1, Byron Unit 2, Braidwood Unit 1, and Braidwood Unit 2). The PRA model is built with a common one-top fault tree, including individual basic events for both Unit 1 and Unit 2 components. The vast majority of the components for Byron and Braidwood are the same, so the vast majority of the fault tree represents both units at both sites. Differences that impact the PRA logic are reflected in the combined PRA fault tree and activated by flags to produce site-specific and unit-specific PRA results.

Separate databases exist for Byron and Braidwood to reflect different operating experience at each site. Separate quantifications are performed for each unit by applying unit-specific flags and the appropriate site-specific database, along with site-specific recovery rules. Site-specific, unit-specific cutset results for each unit are produced (i.e., Byron Unit 1, Byron Unit 2, Braidwood Unit 1, and Braidwood Unit 2).

The internal flooding PRA in integrated into the IEPRA model, and similarly reflects plant-specific or unit-specific differences through the use of flag events and site-specific databases.

The Fire-PRA is built to integrate with the IEPRA using this approach. Due to the physical differences at the plants that impact the Fire PRA, separate FRANX files are developed and applied to produce site-specific results.

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

Response: For the FPIE peer review, one peer review was performed which addressed the models for both sites, given the use of one model and flags to allow quantification for each site and each unit at each site.

For the FPRA, separate peer reviews were performed for each site due to differences in fire results as a result of spatial differences in cable routing. All F&Os from each site were addressed with respect to their impact on both sites.

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?).*

Response: Categorization will be performed using the site- and unit-specific importance results.

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.*

Response: Yes, the current internal events and fire PRA models include credit for the Westinghouse Generation III RCP seals based on PWROG-14001-P, Revision 1 (Reference 6).

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.*

ATTACHMENT 1

Response to Request for Additional Information

Response: Limitations and conditions from the SER that may impact the current PRA model are items 2, 4, & 5.

For item 2, where the identified conditions might occur, the current PRA model accounts for it by treating such a condition as a failure of the shutdown seals. For item 4, the additional failure contribution of the SDS Bypass failure mode will be added to the model that will be used for the 10 CFR 50.69 risk categorization process. For item 5, plant-specific human error probabilities for both of those requirements exist in the current model.

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

Response: No additional exceptions to the limits or conditions exist that may impact applications.

> *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.*

Response: The Generation III Westinghouse RCP seal model did not exist in the plants or in the internal events model that was subject to peer review in 2013.

> *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.*

Response: The peer-reviewed model did include an RCP seal leakage model (WOG 2000) to assess the plant response to events that result from a total loss of cooling to the RCP seals. Implementation of the Generation III Westinghouse RCP seal model was performed consistent with the existing peer-reviewed PRA methods already in the model for the WOG 2000 model.

The change in the seal leakage model is not a new methodology because the new seal leakage model is simply an expansion of the current peer-reviewed model with additional failure probabilities and associated human actions. There is no change in the model scope because the equipment, dependencies, and types of accident sequences remain the same. Finally, there is no change in PRA modeling capability; that is, the peer reviewed PRA model can still evaluate the risk associated with station blackout and total loss of cooling events related to RCP seal failures. Therefore, implementation of the new seal leakage model is a change implemented within the framework of the existing peer-reviewed PRA model structure.

The addition of the Generation III Westinghouse RCP seal model is only a change in the expected seal leakages associated with the new seals. The framework of the model remains essentially the same, and the High Level and Supporting Requirements (HLRs) in the PRA Standard for the Technical Elements associated with RCP seal modeling (e.g., those within the Accident Sequence Analysis, Data Analysis, Human Reliability Analysis, and Quantification technical elements) will continue to be Met or Not Met regardless of addition of the shutdown seal model. Although the lower seal failure rates affect the ordering of the associated accident sequences and reduce CDF and LERF overall, the associated sequences were not significantly changed and new sequences that were not already modeled in the PRA and peer-reviewed were not generated.

In addition, the F&O Closure review specifically examined the updated seal model as a potential upgrade that would require a focused-scope peer review with the following conclusion:

"While changing RCP Seals is potentially a significant change in scope that impacts significant accident sequences, the post-modification sequences and cutsets were already in the model and only show changes in importance due to deleting sequences and cutsets that are no longer risk-significant. This change process occurs in a Fire PRA during refinement, and does not require a focused-scope Peer Review and thus this is PRA Maintenance.

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.*

Response: Implementation of the new RCP seal model is not an upgrade, per the discussion in item iv above.

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.*

Response: The PRA models do include the Westinghouse Generation III RCP seals as discussed in the responses above.

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.*

Response: Not applicable per the above discussion.

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).

Response: The items identified in the table below are required to be completed prior to implementation of the 10 CFR 50.69 risk categorization process. All items identified below are potential items for a license condition.

ATTACHMENT 1

Response to Request for Additional Information

Associated RAI Response	Description
3.a	The internal events and fire PRA models will be updated to versions that include the updated HVAC modeling prior to implementation of the 10 CFR 50.69 risk categorization process.
3.b	Where breaker coordination could not be confirmed for a unit, the applicable model is being updated so that load side cables are designated as causing the loss of the associated power supply. The FPRA models for Byron and Braidwood will be updated to incorporate failures required to account for instances where breaker coordination cannot be confirmed prior to implementation of the 10 CFR 50.69 risk categorization process.
3.c	To ensure that the impact of the CCDP and CLERP scaling factor adjustments is accounted for in the categorization process, a Fire PRA sensitivity in addition to the sensitivities required by NEI 00-04 Table 5-3 will be performed. If the Fire PRA is updated in the future to eliminate the scaling factor adjustment, this sensitivity calculation would no longer be required.
3.d	Identification of all wall mounted panel configurations with four or more switches will be completed and any resulting changes to the Byron and Braidwood FPRA models to incorporate the impact of these panels will be made prior to implementation of the 10 CFR 50.69 risk categorization process.
3.e	The Byron and Braidwood FPRA models that will be used for 10 CFR 50.69 implementation will include a new sump clogging value consistent with the WCAP-16362-NP guidance.
8.c	The Byron and Braidwood Fire PRAs to be used to support the implementation of the 50.69 categorization will retain a 1E-06 joint HEP floor value and justification will be included in the Fire PRA documentation for specific HEP combinations for which a value of less than 1E-05 is used.
11	The additional failure contribution of the Westinghouse RCP Shutdown Seal Bypass failure mode will be added to the Byron and Braidwood Internal Events and Fire PRA models prior to implementation of the 10 CFR 50.69 risk categorization process.

References

1. Letter from David M. Gullott (Exelon Generation Company, LLC) to U.S. NRC, "Application to Adopt 10 CFR 50.69, 'Risk-informed categorization and treatment of structures, systems, and components for nuclear power reactors'," dated September 1, 2017
2. Letter from David M. Gullott (Exelon Generation Company, LLC) to U.S. NRC, "Correction to License Amendment Request to Adopt 10 CFR 50.69, 'Risk-informed categorization and treatment of structures, systems, and components for nuclear power reactors'," dated April 4, 2018

ATTACHMENT 1

Response to Request for Additional Information

3. Email from J. Wiebe (U.S. NRC) to R. Sprengel (Exelon Generation Company, LLC), "Preliminary RAIs for Braidwood and Byron Stations 50.69 Amendment," dated May 9, 2018
4. "PRA Finding Level Fact And Observation Technical Review Of Byron And Braidwood Nuclear Power Plants Units 1 and 2", Revision 2, Jensen Hughes Report 032299-RPT-05, May 2018
5. BAR/BWAR-0-31/34-A1/A2, Byron and Braidwood EDG room cooling alarm response procedures
6. PWROG-14001-P, Revision 1, "PRA Model for the Generation III Westinghouse Shutdown Seal" and associated NRC Safety Evaluation (ADAMS Package Accession No. ML17200A116)