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10 CFR 50.90

May 7, 2018

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

Peach Bottom Atomic Power Station, Units 2 and 3
Renewed Facility Operating License Nos. DPR-44 and DPR-56
NRC Docket Nos. 50-277 and 50-278

Subject: Response to Request for Additional Information
Application to Adopt 10 CFR 50.69, "Risk-informed categorization and treatment of structures, systems, and components for nuclear power plants"

References:

1. Letter from James Barstow (Exelon Generation Company, LLC) to U.S. Nuclear Regulatory Commission – "Application to Adopt 10 CFR 50.69, 'Risk-informed categorization and treatment of structures, systems, and components for nuclear power plants,'" dated August 30, 2017 (ML17243A014)

2. Letter from Richard B. Ennis (U.S. Nuclear Regulatory Commission) to Bryan C. Hanson, Exelon Generation Company, LLC – "Supplemental Information Needed for Acceptance of Requested Licensing Action Re: Adoption of title 10 of the Code of Federal Regulations Section 50.69," dated October 10, 2017 (ML17272B016)

3. Letter from James Barstow (Exelon Generation Company, LLC) to U.S. Nuclear Regulatory Commission – "Supplement to Application to Adopt 10 CFR 50.69, Risk-Informed Categorization and Treatment of Structures, Systems, and Components for Nuclear Power Reactors," dated October 24, 2017 (ML17297B521)

4. Electronic mail message from Jennifer Tobin (U.S. Nuclear Regulatory Commission) to David Helker, Exelon Generation Company, LLC – "Draft 50.69 Request for Additional Information (RAIs) - Peach Bottom," dated March 21, 2018

5. Electronic mail message from Jennifer Tobin (U.S. Nuclear Regulatory Commission) to David Helker, Exelon Generation Company, LLC – "Peach Bottom Units 2 and 3 - Request for Additional Information - Adopt 50.69 License Amendment," dated April 6, 2018

By letter dated August 30, 2017 (Reference 1), Exelon Generation Company, LLC (Exelon) requested an amendment to the Renewed Facility Operating License Nos. DPR-44 and DPR-56 for Peach Bottom Atomic Power Station (PBAPS), Units 2 and 3, respectively. The proposed amendment would modify the licensing basis by the addition of a license condition to allow for the implementation of the provisions of Title 10 of the Code of Federal Regulations (10 CFR), Part 50.69, "Risk-informed categorization and treatment of structures, systems, and components for nuclear power reactors." In a letter dated October 10, 2017 (Reference 2), the U.S. Nuclear Regulatory Commission (NRC) requested that Exelon provide supplemental information in support of the application. By letter dated October 24, 2017 (Reference 3), Exelon responded to the NRC's request for supplemental information.

The NRC reviewed the information provided in the Reference 1 and 2 submittals and indicated the need for additional information in order to complete their review and evaluation of the amendment request. In an electronic mail message dated March 21, 2018 (Reference 4), the NRC issued a draft Request for Additional Information (RAI). This draft RAI was the subject of further discussions during a teleconference on April 6, 2018, between Exelon and NRC representatives and additional clarification was provided. The NRC then issued the formal RAI (Reference 5) in which they requested a response by May 7, 2018.

Attachment 1 to this letter provides a restatement of the RAI questions followed by Exelon's responses. Attachment 2 contains a list of 50.69 Probabilistic Risk Assessment (PRA) implementation items. Attachment 3 contains the proposed markups of the PBAPS, Units 2 and 3, Renewed Facility Operating Licenses.

Exelon has reviewed the information supporting a finding of no significant hazards consideration, and the environmental consideration, that were previously provided to the NRC in Attachment 1 of the Reference 1 letter. Exelon has concluded that the information provided in this response does not affect the bases for concluding that the proposed license amendment does not involve a significant hazards consideration under the standards set forth in 10 CFR 50.92. In addition, Exelon has concluded that the information in this response does not affect the bases for concluding that neither an environmental impact statement nor an environmental assessment needs to be prepared in connection with the proposed amendment.

There are no regulatory commitments in this response.

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

If you have any questions or require additional information, please contact Richard Gropp at 610-765-5557.

I declare under penalty of perjury that the foregoing is true and correct. Executed on the 7th day of May 2018.

Respectfully,

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

Attachments:

1. Response to Request for Additional Information, Application to Adopt 10 CFR 50.69, "Risk-informed categorization and treatment of structures, systems, and components for nuclear power plants"
2. Peach Bottom 50.69 PRA Implementation Items
3. Markup of Proposed Renewed Facility Operating License (RFOL) Pages

cc: w/ Attachments
Regional Administrator - NRC Region I
NRC Senior Resident Inspector - Peach Bottom Atomic Power Station
NRC Project Manager, NRR - Peach Bottom Atomic Power Station
Director, Bureau of Radiation Protection – Pennsylvania Department of Environmental Protection
R.R. Janati, Pennsylvania Bureau of Radiation Protection
S.T. Gray, State of Maryland

ATTACHMENT 1

License Amendment Request

Peach Bottom Atomic Power Station, Units 2 and 3
Docket Nos. 50-277 and 50-278

Response to Request for Additional Information
Application to Adopt 10 CFR 50.69, "Risk-informed categorization and treatment of structures, systems, and components for nuclear power plants"

Attachment 1

Response to Request for Additional Information
Application to Adopt 10 CFR 50.69, "Risk-informed categorization and treatment of structures, systems, and components for nuclear power plants"

By letter dated August 30, 2017 (Reference 1), Exelon Generation Company, LLC (Exelon) requested an amendment to the Renewed Facility Operating License Nos. DPR-44 and DPR-56 for Peach Bottom Atomic Power Station (PBAPS), Units 2 and 3, respectively. The proposed amendment would modify the licensing basis by the addition of a license condition to allow for the implementation of the provisions of Title 10 of the Code of Federal Regulations (10 CFR), Part 50.69, "Risk-informed categorization and treatment of structures, systems, and components for nuclear power reactors." In a letter dated October 10, 2017 (Reference 2), the U.S. Nuclear Regulatory Commission (NRC) requested that Exelon provide supplemental information in support of the application. By letter dated October 24, 2017 (Reference 3), Exelon responded to the NRC's request for supplemental information.

The NRC reviewed the information provided in the Reference 1 and 2 submittals and indicated the need for additional information in order to complete their review and evaluation of the amendment request. In an electronic mail message dated March 21, 2018 (Reference 4), the NRC issued a draft Request for Additional Information (RAI). This draft RAI was the subject of further discussions during a teleconference on April 6, 2018, between Exelon and NRC representatives and additional clarification was provided. The NRC then issued the formal RAI (Reference 5) in which they requested a response by May 7, 2018.

Below is a restatement of the questions followed by Exelon's responses.

RAI 01 – Scope and Quality of PRA Self-Assessments and Peer Reviews

Section 50.69(c)(i) of 10 CFR 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 licensee conducted a self-assessment and a full-scope peer review for its internal events PRA (IEPRA) and fire PRA (FPRA) in November 2010 and 2011, respectively. However, the licensee did not indicate what standard the PRAs were reviewed against or whether internal flooding was included in the scope of review. Address the following:

a) Describe the scope and reason for the self-assessments and peer reviews performed for the IEPRA and FPRA, and confirm they were performed against the guidance in RG 1.200, Revision 2.

Response:

Per the Full Power Internal Events (FPIE) and Internal Flood (IF) PRA Peer Review report (Reference 6) that was signed off in 2011:

"The Peach Bottom Atomic Power Station (PBAPS) Probabilistic Risk Assessment

(PRA) Peer Review was performed in November 2010 at the Exelon offices in Kennett Square, PA, using the NEI 05-04 process, the ASME PRA Standard (ASME/ANS RA-Sa-2009) and Regulatory Guide 1.200, Rev. 2. The purpose of this review was to provide a method for establishing the technical adequacy of the PRA for the spectrum of potential risk-informed plant licensing applications for which the PRA may be used. The 2010 PBAPS PRA Peer Review was a full-scope review of the Technical Elements of the internal events and internal flood, at-power PRA."

Per the FPRA Peer Review report (Reference 7) that was signed off in 2013:

"The Peach Bottom (PB) Atomic Power Station Unit 2 Fire Probabilistic Risk Assessment (FPRA) Peer Review was performed December 3 to 7, 2012 at the Exelon Main Offices in Kennett Square using the NEI 07-12 process, the ASME PRA Standard (ASME/ANS RA-Sa-2009) and Regulatory Guide 1.200, Rev. 2. The purpose of this review was to establish the technical adequacy of the FPRA for the spectrum of potential risk-informed plant licensing applications for which the FPRA may be used. The 2012 PB FPRA Peer Review was a full-scope review of all of the technical elements of the PB at-power Fire PRA (2012 Model of Record) against all technical elements in Section 4 of the ASME/ANS Combined PRA Standard, including the referenced internal events Supporting Requirements (SRs) in Section 2."

Self-assessments against earlier available versions of the PRA Standard and Regulatory Guide (RG) 1.200 were performed over time prior to the FPIE and IF PRA Peer Review, to establish and maintain the technical adequacy and plant fidelity of the PRA model.

b) Confirm that the IEPRA full-scope peer review included internal flooding or otherwise provide the history of peer reviews for the internal flooding PRA.

Response:

Yes, the FPIE Peer Review included a review of the IF PRA model and documentation.

Also, see response to item a) above.

c) Identify which findings presented in Attachment 3 of the LAR were self-assessment findings and which were full-scope peer review findings.

Response:

All findings presented in Attachment 3 of the LAR (Attachment 1) were either full-scope peer review or focused-scope peer review findings. None of the findings are associated with the self-assessment.

The F&Os numbered with a "2011" prefix were originated as part of the Internal Events and Internal Flooding PRA full-scope peer review (Reference 6), those numbered with a "2012" prefix were originated from the Fire PRA full-scope peer review (Reference 7), and those numbered with a "2016" prefix were originated as a part of the Fire PRA focused-scope peer review (Reference 8).

RAI 02 – PRA Upgrades

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 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). In the supplement dated October 24, 2017, the licensee stated that the F&O closure review team performed an assessment of whether each finding resolution constitutes a PRA "upgrade" and identified one upgrade. In addition, the licensee stated that the independent peer review team did consider whether any other of the finding resolutions should have been considered an upgrade and that the review team concurred with the assessment that there were no other upgrades. However, the licensee did not discuss what the upgrade was or how review of the upgrade was completed.

Describe the change identified as an "upgrade," provide a description of the focused-scope peer review performed on the upgrade, and include all resulting F&Os with complete disposition for the application.

Response:

The change identified as an "upgrade" was the enhancement to the fire modeling calculations to incorporate the Thermally-Induced Electrical Failure (THIEF) model (NUREG/CR-6931, Volume 3, (Reference 9)), Flame Spread over Horizontal Cable Trays (FLASH-CAT) model (NUREG/CR-7010, Volume 1, (Reference 10)), and time to automatic detection calculations. The upgrade affects three High Level Requirements (HLRs) under the Fire Scenario Selection Technical Element (TE), FSS-C, FSS-D, and FSS-H. Therefore, the focused scope review addressed the Supporting Requirements (SRs) under these HLRs. The F&Os numbered with a "2016" prefix included in the License Amendment Request (LAR) submittal were originated as a part of this focused-scope peer review.

RAI 03 – Open F&Os in the Process of Being Resolved

Section 50.69(c)(i) of 10 CFR 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 provides F&Os and self-assessment findings that are still open or partially resolved following the F&O closure review. The dispositions for several of these F&Os do not contain sufficient information to justify the licensee's conclusion that the open F&O has minimal or no impact on the application. The specific F&Os and requested information are as follows:

a) Internal events F&Os 2011-3-1, 3-4, 3-6, and 5-8 pertaining to modeling of pre-initiators. These F&Os and their corresponding dispositions state that the licensee did not derive the currently modeled pre-initiators from a formal review of procedures and practices at the plant. The licensee stated that a resolution of these F&Os would have

a minimal impact on this application, and are therefore not necessary. Provide the following:

i. A justification clearly showing why not resolving F&Os 2011-3-1, 3-4, 3-6, and 5-8 has no impact on the 10 CFR 50.69 categorization results, or

Response:

See response to item ii below.

ii. A mechanism that ensures a review of the procedures and practices at the plant is conducted, and that any pre-initiators identified from the review are included in the PRA models prior to implementing the 10 CFR 50.69 categorization process.

Response:

The pre-initiator analysis is currently being refined. The updated evaluation will be incorporated into the 2018 FPIE PRA Update and will resolve F&Os 2011-3-1, 3-4, 3-6, and 5-8. The enhancement to the pre-initiator analysis is considered an Upgrade and therefore a focused scope peer review is planned for this change to ensure the revised pre-initiator analysis meets the supporting requirements of the PRA Standard and that the pre-initiators are implemented into the PRA model appropriately. The 2018 PRA Update and the focused scope peer review of this upgrade will be performed, and any new F&Os resulting from the focused scope review will be resolved, prior to implementation of the 50.69 process.

b) Fire F&O 2012-1-33 pertaining to adjustments to floor-area-ratios. This F&O and the corresponding disposition states that the licensee did not consider correctly the treatment of obstructed floor space and consideration of maintenance practice when determining the floor-area-ratios (FARs) used to allocate the transient ignition frequency across a Physical Analysis Unit (PAU). The F&O indicates that the licensee did not remove the area-obstructed locations from the denominator in the FARs equations. In addition, the disposition indicates that the licensee did not use maintenance practice to inform the likelihood of transient locations. The disposition states that the licensee will resolve the resulting adjustment to the transient fire scenario frequencies in the current model update and that the resolution has minimal impact on this application. Provide the following:

i. A justification clearly showing why not resolving F&O 2012-1-33 prior to implementing the 10 CFR 50.69 categorization process has no impact on the 10 CFR 50.69 categorization results or,

Response:

See response to item ii below.

ii. A mechanism that ensures the FARs are adjusted as discussed in the F&O resolution prior to implementing the 10 CFR 50.69 categorization process.

Response:

This finding will be resolved prior to implementing 10 CFR 50.69. The consideration of maintenance practices to distribute the transient ignition frequency within a PAU is not part of the guidance in NUREG/CR-6850 (Reference 11) and is not required by the ASME/ANS PRA Standard (Reference 12). The transient floor area ratios (FARs) will be adjusted to consider the treatment of obstructed floor space to provide a more accurate distribution of transient ignition frequency. The updated fire PRA will be used for 50.69 categorization.

c) Fire F&O 2012-1-40 pertaining to lack of detailed fire modeling (FM). The F&O cites oversimplified FM in risk significant scenarios. The disposition states that the F&O closure team found that the licensee did perform detailed FM for risk significant scenarios in the modeling update, with some exceptions. Accordingly, the F&O closure team recommended that the licensee apply the detailed two-point FM approach to those exceptions as described in NUREG-6850, "Electric Power Research Institute (EPRI)/NRC-RES Fire PRA Methodology for Nuclear Power Facilities," (ADAMS Accession Nos. ML052580075, ML052580118, and ML103090242) in which models are developed at the 98th and 75th percentiles. The disposition states that the licensee will perform the recommended two-point FM and that the resolution has minimal impact on this application. Provide the following:

i. A justification clearly showing why not resolving F&O 2012-1-40 prior to implementing the 10 CFR 50.69 categorization process has no impact on the 10 CFR 50.69 categorization results or,

Response:

See response to item ii below.

ii. A mechanism that ensures the two-point FM is applied to the risk-significant fire scenarios prior to implementing the 10 CFR 50.69 categorization process.

Response:

This finding will be resolved prior to implementing 10 CFR 50.69. Risk significant scenarios for ignition sources capable of being modeled with a two-point fire modeling approach will be updated to include a two (or more-) point fire modeling approach. The updated fire PRA that models the risk significant scenarios with a two (or more-) point fire modeling approach will be used in the categorization process.

d) Fire F&O 2012-3-17 pertaining to credit for fire wrapping. The disposition to this F&O states that the FPRA currently credits fire wrap for protecting enclosed cables though there may be locations potentially vulnerable to mechanical damage. Accordingly, the disposition states that the licensee will conduct a review of potentially vulnerable fire wrap configurations and, if vulnerabilities are confirmed, then the licensee will remove credit for the fire wrap from the FPRA. Provide the following:

i. A justification clearly showing why not resolving F&O 2012-3-17 prior to implementing the 10 CFR 50.69 categorization process has no impact on the 10 CFR 50.69 categorization results or,

Response:

See response to item ii below.

ii. A mechanism that ensures a review is conducted of potentially vulnerable fire wrap configurations and credit removed from the PRA for any fire wrap susceptible to mechanical damage prior to implementing the 10 CFR 50.69 categorization process.

Response:

This finding will be resolved prior to implementing 10 CFR 50.69. A review will be conducted of potentially vulnerable fire wrap configurations to identify which are subject to mechanical damage. The fire PRA will be updated to ensure that fire wrap configurations are not credited in fire scenarios that could subject fire wrap to mechanical damage. The updated fire PRA that ensures the fire wrap configurations are not credited in scenarios where they would be subject to mechanical damage will be used in the 50.69 categorization process.

e) Fire F&O 2012-3-37 pertaining to review of excluded electrical panels. The F&O states that the licensee did not consider failure of electrical panels and all targets terminating at the panel for all non-propagating electrical panel fires that are modeled in the FPRA. The disposition states that the licensee will confirm that the excluded panels lead to a single failure, and that the licensee will incorporate any excluded panels into the FPRA if the licensee determines that multiple failures from an excluded panel fire are possible. The licensee states that this F&O is being resolved in the current model update and has minimal impact on this application. In addition, this resolution does not appear to be consistent with the NRC guidance in NUREG/CR-6850, "Electric Power Research Institute (EPRI)/NRC-RES Fire PRA Methodology for Nuclear Power Facilities," (ADAMS Accession Nos. ML052580075, ML052580118, and ML103090242), for excluding electrical cabinets from being counted if they are simple wall-mounted panels with less than four switches or they are well-sealed and robustly secured cabinets containing circuits below 440 V.

Explain how the resolution to F&O 2012-3-37 is consistent with the guidance contained in NUREG/CR-6850, or provide a description and justification for the proposed resolution.

Response:

The resolution to F&O 2012-3-37 in the current fire PRA model is consistent with the guidance in NUREG/CR-6850 (Reference 11). Per the guidance, the fire PRA excludes panels as ignition sources if they are simple wall-mounted panels with less than four switches or if they are well-sealed and robustly secured cabinets containing

circuits below 440V. The additional criteria, to confirm that the panel only leads to a single failure, included in our original disposition of the finding is not consistent with NUREG/CR-6850 and will therefore not be included in the fire PRA model.

f) Fire F&O 2012-6-1 pertaining to resolution of internal events findings. The disposition to this F&O states that internal events F&O 6-11 (regarding supporting requirement (SR) DA-C11) and F&O 3-6 (regarding SR HR-A1 and HR-A2) remained open for the FPRA. The F&O states for internal events F&O FPIE 6-11 (mistakenly referred to as F&O 6-1 in PB-PRA-021.01 and the FPIE PR report) that validation is needed for use of maintenance rule unavailability data in the PRA. Internal events F&O 3-6 pertaining to identification of pre-initiators is already addressed in this question in part (a). The licensee states that F&O 2012-6-1 has not been resolved but has minimal impact on the 10 CFR 50.69 categorization process.

i. Confirm that the internal events F&O 6-11 cited in Fire F&O 2012-6-1 was closed by the 2016 F&O Closure process, or alternatively provide description of the F&O and its resolution or disposition for the 10 CFR 50.69 application.

Response:

F&O 6-11 was not closed by the 2016 F&O closure process.

Finding 6-11 Description

The data from the Maintenance Rule is used directly without checking to see if it includes only those maintenance or test activities that could leave the component, train, or system unable to perform its function when demanded as required by the SR.

Disposition for 50.69

Exelon Procedure No. WC-AA-101 provides the following guidelines for how to track component, train, or system unavailability and states the following for MRULE monitoring:

"SSCs out of service for testing are considered unavailable, unless the function can be promptly restored either by an operator in the control room or by a dedicated operator stationed locally for that purpose. Restoration actions must be contained in a written procedure, must be uncomplicated (a single action or a few simple actions), and must not require diagnosis or repair. Credit for a dedicated local operator can be taken only if (s)he is positioned at the proper location throughout the duration of the test for the purpose of restoration of the train should a valid demand occur. The intent of this paragraph is to allow licensees to take credit for restoration actions that are virtually certain to be successful (i.e., probability nearly equal to 1) during accident conditions."

Therefore, using raw data taken directly from the Maintenance Rule provides very realistic testing and maintenance failure probabilities for the PRA model as any testing or maintenance that would not make the SSC unavailable would not be tracked in the Maintenance Rule.

Conclusion

The resolution of F&O 6-11 is a documentation enhancement and therefore will have no impact on either the internal events PRA or the fire PRA. The resolution of this F&O has no impact on the 50.69 categorization process.

The Table provided in response to RAI 14 includes additional details for F&O 6-11.

ii. Provide a justification clearly showing why not resolving F&O 2012-6-1 has no impact on the 10 CFR 50.69 categorization results.

Response:

The two FPIE PRA findings identified in FPRA finding 2012-6-1 that were not considered fully closed by the finding closure review are either documentation related (2011-6-11, see response to item i above, which provides the basis for no impact) or pre-initiator related (2011-3-6, see response to RAI 03.a item ii., which explains that this issue is being resolved in the PRA prior to implementation of 50.69 for PBAPS). Therefore, there is no impact on 50.69 categorization results.

RAI 04 – Open/Partially Open Findings Resolved Using Sensitivity Studies

Section 50.69(c)(i) of 10 CFR 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. The resolutions to a number of F&Os presented in Attachment 3 of the LAR involve sensitivity studies that the licensee will perform to validate the treatments used in the PRA. The dispositions for these cases do not explain how the licensee will resolve the F&Os if the sensitivity study results do not validate that the F&O has only a minimal impact on the application. The specific F&Os and requested information are as follows:

a) Fire F&O 2012-5-6 pertaining to the uncertainty of FM parameters. The disposition for this F&O states that to resolve this F&O the licensee will evaluate FM parameter uncertainties for risk significant scenarios and that the licensee will perform sensitivity studies for parameters identified to be key sources of uncertainty. The licensee states that this F&O is being resolved in the current model update and has minimal impact on this application. Provide the results of the sensitivity studies on the FM parameters to demonstrate no impact on the 10 CFR 50.69 categorization results, or if any impact is identified, describe the actions that will be taken to address it.

Response:

The fire modeling parameters for which there exist uncertainty are associated with the application of the THIEF model (Reference 9), automatic detection model, and FLASH-CAT model (Reference 10) within the fire modeling calculations.

The THIEF model predicts the temperature profile within a cable as a function of time to simulate the delay in damage to a cable. By varying the THIEF input parameters, the impact is limited to either increasing or decreasing the time available to suppress the fire prior to damage. The categorization process includes the sensitivity studies from NEI 00-04, Table 5-3, which includes the sensitivity study to remove all credit for manual suppression. Another sensitivity study will be performed as part of the categorization process which assumes credit for immediate manual suppression. These sensitivities bound the variations that may be identified by varying the input parameters to the THIEF model.

The automatic detection model predicts the time at which the credited automatic detection systems are activated which influences the time available to perform manual suppression. As described above, the categorization process includes the sensitivity studies from NEI 00-04, Table 5-3, which includes the sensitivity study to remove all credit for manual suppression. Another sensitivity study will be performed as part of the categorization process which assumes credit for immediate manual suppression. These sensitivities bound the variations that may be identified by varying the input parameters to the automatic detection model.

The implementation of FLASH-CAT in the peer reviewed fire PRA model introduced sources of uncertainty by using generic parameters from NUREG-7010 Volume 1 (Reference 10) and weighted averages of parameters for cables located within the Physical Analysis Units (PAUs) in which the scenarios implementing the FLASH-CAT model were located. To remove these sources of uncertainty, values used for these parameters (mass per unit length and plastic mass fraction) will be based on the scenario specific set of cables that are located within the cable trays analyzed using the FLASH-CAT model. These changes will be made in the current model update to the fire PRA prior to implementation of 50.69 and will therefore eliminate use of FLASH-CAT in the PRA model as a source of uncertainty in the 50.69 categorization process.

b) Fire F&O 2016-1-1 pertaining to the assumption that control cables would be the critical target cable in most scenarios. The disposition states that to resolve the F&O the licensee will use the weighted average of cable parameters in a PAU for thermally-induced electrical failure (THIEF) calculations (in some cases specific target parameters were used). The disposition states that the licensee is performing sensitivity studies by using the upper and lower bounds of these cable parameters as input to THIEF (for calculating critical time to cable insulation failure) to compare to the results of running THIEF using the mean parameters. The licensee states that this F&O is being resolved in the current model update and has minimal impact on this application. Provide the results of the sensitivity studies on the cable parameters to demonstrate no impact on the 10 CFR 50.69 categorization results, or if any impact is identified, describe the actions that will be taken to address it.

Response:

As described in the response to part a) of RAI 04 above, the THIEF model predicts the temperature profile within a cable as a function of time to simulate the delay in damage to a cable. By varying the THIEF input parameters, the impact is limited to either increasing or decreasing the time available to suppress the fire prior to damage. The categorization

process includes the sensitivity studies from NEI 00-04, Table 5-3, which includes the sensitivity study to remove all credit for manual suppression. Another sensitivity study will be performed as part of the categorization process which assumes credit for immediate manual suppression. These sensitivities bound the variations that may be identified by varying the input parameters to the THIEF model.

c) Fire F&O 2016-1-2 pertaining to the radial increments used in THIEF. The disposition to this F&O states that to resolve this F&O the licensee will perform a sensitivity study on the "radial increment" used in the THIEF modeling, which is larger than the radial increment recommended in NUREG/CR-6931, Volume 3, "Cable Response to Live Fire (CAROLFIRE) Volume 3: Thermally-Induced Electrical Failure (THIEF) Model," (ADAMS Accession No. ML081190261) to demonstrate that the increment used yields a reasonable but conservative result. In addition, the licensee states that this F&O is being resolved in the current model update and has minimal impact on this application. Provide the results of the sensitivity studies on the radial increments to demonstrate no impact on the 10 CFR 50.69 categorization results, or if any impact is identified, describe the actions that will be taken to address it.

Response:

A generic sensitivity was performed using the Electric Power Research Institute (EPRI) Fire-Induced Vulnerability Evaluation (FIVE) THIEF (Reference 24) function in which the radial increments were adjusted for a variation of cable configurations. Radial increments of 0.1, 0.2, 0.3, 0.4, and 0.5 mm were tested to see the impact of changing this parameter. The times calculated for cable damage varied in both directions. The differences in times to failure observed when compared to the times calculated using 0.1 mm ranged from -10 to +13 seconds (-1.35% to 5.83%) across the various cable configurations and radial increments tested. Looking specifically at the 0.5 mm radial increment the differences in times to failure ranged from +2 to +7 seconds (0.27% to 3.14%). Therefore, it can be concluded that increasing the radial increment value from 0.1 mm to 0.5 mm does not represent a significant change to the time to cable failure and therefore the impact on the calculated Non-Suppression Probabilities is also not significant. Any potential impact to the calculated time delay to cable damage would also be bounded by the sensitivity studies performed on manual suppression described in the responses to RAI 04 parts a) and b) above.

d) Fire F&O 2016-1-6 pertaining to the assumed flame spread over horizontal cable trays (FLASH-CAT) and THIEF modeling parameters. The disposition to this F&O states that the licensee will perform sensitivity studies that vary the parameters used in FLASH-CAT and THIEF between reasonable upper and lower bounds in order to determine the impact on analysis. The licensee states that this F&O is being resolved in the current model update and has minimal impact on this application. Provide the results of the sensitivity studies on the fire modeling parameters (both FLASH-CAT and THIEF) to demonstrate no impact on the 10 CFR 50.69 categorization results, or if any impact is identified, describe the actions that will be taken to address it.

Response:

The THIEF model (Reference 9) predicts the temperature profile within a cable as a function of time to simulate the delay in damage to a cable. By varying the THIEF input parameters, the impact is limited to either increasing or decreasing the time available to suppress the fire prior to damage. The categorization process includes the sensitivity studies from NEI 00-04, Table 5-3, which includes the sensitivity study to remove all credit for manual suppression. Another sensitivity study will be performed as part of the categorization process which assumes credit for immediate manual suppression. These sensitivities bound the variations that may be identified by varying the input parameters to the THIEF model.

The implementation of FLASH-CAT (Reference 10) in the peer reviewed fire PRA model introduced sources of uncertainty by using generic parameters from NUREG-7010, Volume 1 (Reference 10) and weighted averages of parameters for cables located within the PAUs in which the scenarios implementing the FLASH-CAT model were located. To remove these sources of uncertainty, values used for these parameters (mass per unit length and plastic mass fraction) will be based on the scenario specific set of cables that are located within the cable trays analyzed using the FLASH-CAT model. These changes will be made in the current model update to the fire PRA prior to implementation of 50.69 and will therefore eliminate use of FLASH-CAT in the PRA model as a source of uncertainty in the 50.69 categorization process.

RAI 05 – Open F&Os with Incomplete Dispositions

Section 50.69(c)(i) of 10 CFR 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. The following dispositions for open or partially resolved F&Os presented in Attachment 3 of the LAR seemed incomplete and require further justification and/or information:

a) Undesired Operator Actions in FPRA

The dispositions for F&Os 2012-2-6 and 2012-2-7, pertaining to undesired operator actions, state that the licensee will perform a procedure-by-procedure review to justify why consideration of instrumentation that could potentially mislead operators or cause them to perform a harmful action did not need to be modelled in the FPRA. The licensee states that if a justification cannot be provided, then the licensee will incorporate the undesired operator actions into the PRA models as applicable. However, the disposition does not explain how the licensee will incorporate these undesired operator actions, which could propagate errors of commission and errors of omission into the PRA model. In addition, the disposition does not indicate whether this update would meet the criteria for a PRA "upgrade" as defined in the PRA Standard (i.e., American Society of Mechanical Engineers/American Nuclear Society (ASME/ANS) RA-Sa-2009, "Addenda to ASME/ANS RA-S-2008, Standard for Level 1/Large Early Release Frequency Probabilistic Risk Assessment for Nuclear Power Plant Applications"), as endorsed by

RG 1.200. Provide the following:

i. A detailed discussion describing how the licensee will incorporate any applicable undesired operator actions into the PRA model. The discussion should include a description of the modeling of failures to respond to fire-induced damage of PRA modeled components (i.e., failure of omission), and undesired actions that could to lead to undesired equipment actuations (i.e., errors of commission).

ii. A discussion of whether the PRA model update to include any applicable undesired operator actions constitutes a PRA "upgrade" as defined in the PRA Standard (i.e., ASME/ANS RA-Sa-2009), as endorsed by RG 1.200. If the update is considered a PRA "upgrade", then propose a mechanism that ensures a focused-scope peer review of the upgrade is performed and all resulting F&Os are resolved prior to implementing the 10 CFR 50.69 categorization process.

Responses to items i and ii above:

The NUREG-1921 (Reference 14) methodology is being applied in identifying undesired operator actions and will be used to incorporate any identified actions into the fire PRA. An update to include any applicable undesired operator actions would be considered an "upgrade" as defined in the PRA Standard, so a focused-scope peer review of the application of this methodology will be performed, and any new F&Os resulting from the focused scope review will be resolved prior to implementation of the 50.69 process.

b) Uncoordinated Breakers in FPRA

The disposition for F&O 2012-5-1 states that circuits modeled in the FPRA determined to be uncoordinated will be modelled as "non-coordinated" in the FPRA. However, the disposition does not describe how the licensee will model the non-coordinated circuits. Failures caused by these inadequate circuits could represent a new kind of failure not currently modeled in the FPRA. The disposition does not indicate whether this update would meet the criteria for a PRA "upgrade" as defined in the PRA Standard (i.e., ASME/ANS RA-Sa-2009), as endorsed in RG 1.200. Provide the following:

i. A discussion of the non-coordinated circuits and the extent of the modeling necessary to address this F&O.

ii. A detailed discussion describing how the licensee will incorporate any non-coordinated circuits in the PRA model. The discussion should include a description of any modeled component failures associated with the uncoordinated circuit, including failures that could occur upstream and downstream of possible fault locations along the circuit.

iii. A discussion of whether the PRA model update to include non-coordinated circuits constitutes a PRA "upgrade" as defined in the PRA Standard (i.e., ASME/ANS RA-Sa-2009), as endorsed by RG 1.200. If the update is considered

a PRA "upgrade", then provide a mechanism that ensures a focused-scope peer review of the upgrade is performed and all resulting F&Os are resolved prior to implementing the 10 CFR 50.69 categorization process.

Responses to i, ii, and iii above:

The scope of circuits for which coordination has not been confirmed is limited to non-safety related power supplies credited in the fire PRA model. The PRA model will be updated to include failure of the power supply when the power cable within the circuits of concern are identified to be damaged by fire scenarios, or additional analysis will be performed to determine that circuits are coordinated. Failure of the power supply will account for the opening of the power supply upstream breaker that may occur due to the potential lack of coordination between it and the downstream breaker associated with the damaged power cable.

The impact to PRA credited equipment dependent on the power cable is already included in the PRA model.

The update to the PRA model to include non-coordinated circuits does not constitute a PRA "upgrade" as it will be performed in a manner that was already subjected to a Peer Review. The PBAPS Fire PRA model Peer Reviewed in 2012 included the modeling of fault propagation that could occur when fire induced failure of the control circuit is considered (per NEI 00-01, MSO 5n (Reference 15)). Modeling the potential lack of coordination is the same except that the additional failure of the control circuit is not required when it is assumed that the circuit is not properly coordinated with the upstream breaker for the power supply.

RAI 06 – Open/Partially Open F&Os Associated with Fire Modeling Codes

Section 50.69(c)(i) of 10 CFR 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. The dispositions to F&Os 2016-1-3, 2016-1-4, and 2016-1-9, indicate that the findings will be resolved by showing that the THIEF, FLASH-CAT, and Time-to-Automatic-Detection models and related model parameters have been applied within their known limits of applicability. Provide demonstration that the THIEF, FLASH-CAT, and Time-to-Automatic-Detection models and related model parameters have been used within known limits of their applicability. For each of the F&Os 2016-1-3, 2016-1-4, and 2016-1-9, provide a complete description of the F&O, and a discussion of its resolution.

Response:

Finding 2016-1-3 stated that there was no basis documented that the THIEF model is used within its validation range and that this was significant because use of the model outside the validation ranged may lead to inaccurate and potential non-conservative risk estimates. The following discussion provides the basis that the THIEF model was used appropriately. As

documented in NUREG/CR-6931, Volume 3 (Reference 9) the THIEF model was experimentally tested with cables with a range of properties: diameter 7.0-19.0 mm, jacket thickness 0.5-1.5 mm, mass/length 0.076-0.529 kg/m. The ranges of parameters were not identified by NUREG/CR-6931 as limits of applicability. Rather, NUREG/CR-6931 states that: "the range in cable properties demonstrates that the THIEF model is applicable to a wide variety of cables with no need for additional information beyond the cable diameter, mass per length, and an empirical 'failure' temperature. In addition, there was no indication from the model results that indicated a bias related to the number of conductors, plastic composition, or copper content." Furthermore, NUREG-1805, Supplement 1 (Reference 22) includes the THIEF model and considers cables with the following property ranges: diameter 0.912-79.121 mm, jacket thickness 0.0-4.318 mm, mass/length 0.012-17.48 kg/m. This supports the conclusion that the ranges specified in NUREG/CR-6931 are not limits of applicability and that the THIEF model can be applied at PBAPS using the applicable cable properties for the configurations under consideration.

Finding 2016-1-4 stated that the time-to-automatic-detection model was not shown to be used within the known limits of applicability. The recommendation to resolve the finding was to add the ceiling jet distance ratio to the set of non-dimensional parameters to ensure the correlation is used within the known limits for each scenario. To resolve this finding the ceiling jet distance ratio was calculated for the configurations in which the automatic detection model was used. There are cases in which the ratio is outside the validation range included in NUREG-1824, Volume 1 (Reference 16). While this introduces a source of uncertainty to the analysis, the impact is limited to changes to the calculated time to detection, which in turn influences the time available to suppress the fire. The categorization process includes the sensitivity studies from NEI 00-04, Table 5-3, which includes the sensitivity study to remove all credit for manual suppression. Another sensitivity study will be performed as part of the categorization process which assumes credit for immediate manual suppression. These sensitivities bound the uncertainty introduced by using the automatic detection model outside of the ceiling jet ratio validated range.

Finding 2016-1-9 stated that a verification was not performed to ensure that the FLASH-CAT and THIEF models were incorporated properly into the Microsoft (MS) Excel files used to perform fire modeling calculations. To resolve this finding a verification of the FLASH-CAT and THIEF model implementation in the MS Excel Fire Modeling Workbook (FMW) has been performed. The implementation of THIEF was confirmed by testing six (6) cases in the Fire Dynamics Tools (FDTs), EPRI FIVE, and the FMW. The results obtained using the FMW matched the results obtained using EPRI FIVE with the one exception where the FMW calculated the time to failure at 1100 seconds while the FDTs calculated 4000 seconds and EPRI FIVE indicated there would be no failure. The verification of FLASH-CAT was performed with 3 different cable tray configurations against EPRI FIVE. The results of the verification showed that the FMW would slightly overestimate the Heat Release Rate (HRR) produced by secondary combustible cable trays during the growth phase of the fires but that once steady-state conditions were reached the HRRs were nominally the same with a difference of <1% identified. Therefore, verification that the models were incorporated in an appropriate manner has been confirmed.

RAI 07 – Implementation Items

Section 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. Attachment 6 of the LAR presents dispositions for assumptions and modeling uncertainties that include planned updates to the PRA models after the NRC approves the 10 CFR 50.69 implementation amendment request but before the licensee implements its 10 CFR 50.69 categorization program. These updates include updating the internal flooding pipe break frequencies, and removing credit for core melt arrest in vessel. However, the licensee did not discuss whether these future updates can have an impact on the 10 CFR 50.69 categorization process.

a) Justify why the updates included in Attachment 6 of the LAR (i.e., updating the internal flooding pipe break frequencies, and removing credit for core melt arrest in vessel) have no impact on the 10 CFR 50.69 categorization results or include them in response to item b. below.

Response:

Attachment 2 of this response identifies those items in Attachment 6 of the PBAPS 50.69 LAR that are required to be completed prior to implementation of the 10 CFR 50.69 risk categorization process. All issues identified in the attachment will be addressed and any associated changes will be made, focused-scope peer reviews will be performed on changes that are PRA upgrades as defined in the PRA standard (ASME/ANS RA-Sa-2009, as endorsed by RG 1.200, Revision 2), and any findings will be resolved and reflected in the PRA of record prior to implementation of the 10 CFR 50.69 categorization process.

b) Provide a list of activities and 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

Based on the response to part a) above, Exelon proposes to add the following license condition to Appendix C, of the Renewed Facility Operating License Nos. DPR-44 and DPR-56 for PBAPS, Units 2 and 3, respectively (see Attachment 3):

> Exelon is approved to implement 10 CFR 50.69 using the processes for categorization of Risk-Informed Safety Class (RISC)-1, RISC-2, RISC-3, and RISC-4 structures, systems, and components (SSCs) specified in the license amendment dated [DATE] subject to the following condition:
>
> Exelon will complete the implementation items listed in Attachment 2 of Exelon

letter to NRC dated May 7, 2018, prior to implementation of 10 CFR 50.69. All issues identified in the attachment will be addressed and any associated changes will be made, focused-scope peer reviews will be performed on changes that are PRA upgrades as defined in the PRA standard (ASME/ANS RA-Sa-2009, as endorsed by RG 1.200, Revision 2), and any findings will be resolved and reflected in the PRA of record prior to implementation of the 10 CFR 50.69 categorization process.

Prior NRC approval, under 10 CFR 50.90, is required for a change to the categorization process specified above (e.g., change from a seismic margins approach to a seismic probabilistic risk assessment approach).

RAI 08 – Key Assumptions and Uncertainties that could Impact the Application

Sections 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 they do not represent a key source of uncertainty and will not be an issue for the 10 CFR 50.69 categorization. However, in a number of instances the licensee did not provide sufficient information for NRC to conclude that the uncertainty will not impact the 10 CFR 50.69 categorization results. The specific F&Os and requested information are as follows:

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

Attachment 6 (page 71) of the LAR states that the outdoor air temperature at the plant does not often exceed the design basis temperature for the diesel generators (DG) (i.e., >80° F) which would require two DG cooling fans. The licensee states that it assumes just one-of-two DG cooling fans to be an adequate success criterion in the PRA for the entire year. Based on publically available information, the average high temperatures at the plant location appears to be above 80° F in the summer months. Provide the following:

i. A basis for the adequacy of the assumed one-out-of-two DG cooling fans success criteria in the summer months, or

<u>Response:</u>

See response to item iii below.

ii. Alternatively, justify quantitatively that the 10 CFR 50.69 categorization results are not sensitive to the assumption that one-of-two DG cooling fans provide adequate cooling when the temperature at the plant exceeds the design basis temperature (i.e., >80° F).

Response:

See response to item iii below.

iii. A mechanism that ensures the model is adjusted to account for higher summer temperatures prior to implementing the 10 CFR 50.69 categorization process.

Response

Prior to implementation of the 50.69 program, the PRA model will be updated to model the requirement of two Emergency Diesel Generator (EDG) cooling fans for the fraction of the year that the average high outdoor temperatures at Peach Bottom are above 80° F.

To incorporate this change into the PRA model, weather information applicable to the site will be obtained and used to determine the amount of time in recent history that the outdoor air temperature exceeds the design basis temperature for the EDGs (which corresponds to outdoor temperature > 80° F). Using this information, the EDG HVAC system logic model will be revised to require both EDG fans when the design basis temperature is exceeded, and only require one EDG fan when the temperature is not exceeded. This enhancement will be incorporated into the logic model for each EDG fan prior to implementation of the 50.69 program. This change does not require the use of any methods not already peer reviewed for the PBAPS PRA and will not result in any new accident sequences. Therefore, this change does not constitute a model upgrade.

b) Low Pressure Injection after Core Damage

Attachment 6 (page 74) of the LAR indicates that the licensee takes credit in the PRA for "timely" low pressure injection after core damage to avoid a large early release. The disposition states that this assumption provides a reasonable best-estimate approach. However, the licensee does not provide a basis for the assumption. Provide the following:

i. A discussion which describes the thermohydraulic basis or rationale supporting the validity of the assumption that injection of low pressure cooling after core damage can prevent a large early release. The discussion should include a description of the term "timely low pressure injection."

Response:

The MAAP4 (Reference 17) computer code is routinely used to simulate the thermohydraulic response for a plant given a set of initiating events and operator actions. It is used at PBAPS to determine timings for both Level 1 (CDF) and Level 2 (release frequency) analyses. Sensitivity cases for PBAPS were investigated using MAAP4 on a scenario that progressed to vessel failure. These cases allowed for emergency depressurization and the restoration of 1 train of LPCI after core damage (assumed to correspond to MAAP4-predicted core exit thermocouple temperature greater than 1800 °F), but before vessel failure.

The sensitivity cases show that the introduction of water into the vessel and the recovering of the core modifies the conditions in the RPV such that the failure mechanism is not able to progress.

"Timely low pressure injection" is therefore defined as restoring LPCI and initiating depressurization at least 30 minutes before vessel failure to support injection. The MAAP cases run to support this RAI response will be incorporated into the Level 2 T/H analysis documentation as a part of the ongoing FPIE PRA Update. There is no change to the model as a result of this documentation enhancement.

ii. Alternatively, justify quantitatively that the 10 CFR 50.69 categorization results are not sensitive to this assumption.

Response:

See response to item i above.

c) Safety Relief Valve (SRV) Success Criteria after Passing Liquid

Attachment 6 (page 75) of the LAR states that although the SRVs are designed to pass water, and 10 CFR Part 50, Appendix R, "Fire protection program for nuclear power facilities operating prior to January 1, 1979," models the RPV being flooded with water returning to the suppression pool via the SRVs, the SRVs are never tested in this fashion. The disposition also states that the licensee assigns a nominal failure probability to the SRVs for failure to ***open*** following flooding of the steam lines that causes liquid to pass through the SRVs. However, the NRC staff notes that the NUREG/CR-6928, "Industry Average Performance for Components and Initiating Events at U.S. Commercial Nuclear Power Plants," (ADAMS Accession No. ML070650650) Component Reliability 2010 update provides a probability estimate for an SRV to fail to ***close*** (passing liquid) of 0.1. Provide the following:

i. Confirm that the nominal failure probability used for the SRVs to fail to open is consistent with the 2010 update of NUREG/CR-6928 or justify the nominal failure probability used, including a justification for assuming the SRVs will work absent of testing.

Response:

Controlled Depressurization Response in the PRA:

The independent failure rate for a single SRV failing to open is 2.77E-03 / demand as provided by the NUREG/CR-6928 (Reference 18) 2010 generic data release. This failure rate is used to calculate the failure probability for the SRVs failing to open basic events in the PRA model. At PBAPS, controlled reactor depressurization will be successful if at least two of 11 safety/relief valves open and remain open. There is a Common Cause Failure (CCF) basic event (ARV--COMCPI2) modeled for 10 of 11 SRVs failing to open which has a probability value of 1.9E-6. The CCF basic event is calculated using the independent failure rate provided by the NUREG/CR-6928 2010

generic data release and using the alpha factor provided by the Idaho National Laboratories (INL) CCF Parameter Estimations, 2012 Update (Reference 23).

Steam Line Flooding Scenario in the PRA:

The steam lines in BWRs may become flooded if level is not maintained below Level 8, the automatic trip functions fail, and operators do not respond in time to take manual control of HPCI or RCIC after the Level 8 trip failure. The water in the steam lines could then disable the SRVs from being able to perform their pressure control function even if the RPV water level drops later.

Failure of the HPCI, RCIC or FW systems to trip on high reactor level is included in the PRA model as leading to a steam line flooding scenario. Given the conditions occur that would allow initial flooding of the steam lines, a probability is assigned that this flooding permanently disables all of the SRVs. This is modeled with basic event APH--SRXDXI2.

Per the PBAPS PRA Data Notebook:

The passing of water through the SRVs temporarily (i.e. should Level 8 trips fail and RPV water level rises to the main steam lines) should not render the SRVs totally unavailable. However, to account for this possibility, a failure probability of 1.0E-3 is assigned to this occurring (APH--SRXDXI2) given that conditions arise to allow water ingress into the main steam lines.

Modeling all SRVs as unavailable given water in the steam lines with a 1.0E-03 failure probability (i.e., an increase by more than a factor of 500 compared to the CCF failure to open basic event ARV--COMCPI2 probability) is considered to be a reasonable representation given the large number of SRVs (11) relative to the number required (2).

ii. Given that the licensee has identified a scenario where SRVs could potentially pass liquid following steam-line flooding, discuss how the licensee addresses the failure of the SRVs to close after passing liquid. The discussion should include consequences of the failure, how/if the failure is modeled, and a justification for the failure to close probabilities used, including a justification for the assumption that the SRVs will work absent of testing, or

Response:

The scenario of the SRVs failing to close given that they are demanded and pass liquid is a potentially beneficial failure that is not currently included in the model. Given water in the steam lines, an increased likelihood of high pressure injection failure is assigned, and depressurization to allow for low pressure injection would be required in those cases. As described above, basic event APH--SRXDXI2 models the failure of all SRVs to depressurize given that water is in the main steam lines. If depressurization were to occur, then a myriad of low pressure injection systems are available to avoid core damage. Taking credit for this beneficial failure in these scenarios would result in a further reduction to the APH--SRXDXI value utilized in the model.

iii. As an alternative to ii., justify, quantitatively, that the 10 CFR 50.69 categorization results are not sensitive to the assumption or SRV failure modes addressed in items i and ii above.

Response:

See responses to items i and ii above.

d) Turbine success while passing liquid

Attachment 6 (page 76) of the LAR states that a steam turbine similar to the reactor core isolation cooling (RCIC) and high-pressure coolant injection (HPCI) turbines was tested with a slug of water and found to run successfully. The LAR states that based on this information, the licensee assigned a nominal failure to provide a slight conservative bias for this failure mode. However, the licensee does not provide a justification showing why a single test on another turbine is sufficient to support the assumption that RCIC and HPCI turbines may continue running after passing liquid. In addition, the licensee does not provide information describing what nominal failure probability value the licensee assigned to this failure mode and why the licensee considered the nominal probability to be conservative. Provide the following:

i. A discussion describing whether the RCIC and HPCI turbines were designed to continue running after passing liquid. If the RCIC and HPCI turbines were designed to continue running while and after passing liquid, then discuss whether these conditions encompass the conditions that would occur in the accident scenarios for which these turbines were credited (which appears to include instances in which there is continuous flow of water for the scenario).

Response:

Failure of the HPCI, RCIC or Feedwater (FW) systems to trip on high reactor level (Level 8) is included in the PRA model as potentially leading to a steam line flooding scenario. Given the high reactor water level trips fail, initial flooding of the steam lines could occur. The operator response times to prevent this from occurring are based on the RPV water level initially reaching the main steam line elevation. Given this occurs, a probability is assigned that this could lead to complete failure of the HPCI and RCIC turbines. Note that the FW turbines are assumed to be failed in these same scenarios.

Although HPCI and RCIC are not specifically designed to continue running while passing liquid, information from the HPCI and RCIC turbine manufacturer, Terry Steam Turbine Division - Terry Corporation (Terry), provides a basis for concluding that the turbine design used in the PBAPS HPCI and RCIC systems will operate under water ingestion conditions. Terry Corporation Bulletin No. S-213 (Reference 25) summarizes a series of tests performed on this type of turbine. Reference 26 is a letter from Terry to General Electric Company (GE), the PBAPS nuclear steam supply system (NSSS) designer, attesting to the ability of the HPCI turbines that Terry supplied to GE for a number of GE BWRs based on test results as summarized in

Exhibit 1. Reference 27 is a letter from Terry to GE confirming that the results of the HPCI turbine water ingestion test are applicable to the turbines used for RCIC as supplied to GE for a number of GE BWRs, and attesting to RCIC operation under these conditions. Although the specific PBAPS units have not been confirmed to be listed on these letters, the PBAPS HPCI and RCIC turbines are of designs common to those supplied by Terry Corporation for most GE BWRs. Note that References 25 through 27 are not the property of Exelon and should the NRC need to review this documentation in support of its review of this LAR, the information can be made available via an electronic portal (e.g., eDocs).

Further, in the scenarios of interest the water ingress into the turbine would be somewhat self-limiting (i.e., HPCI and RCIC would not be demanded until after FW fails and the RPV water level recedes, or a two-phase mixture would likely slow the turbines down, initially decreasing the flow from HPCI and/or RCIC), and therefore would not totally disable the pumps from running.

This is likely what happened at Fukushima Unit 2 where RCIC continued to inject without any level control for 67 hours before failing. Per SANDIA Report SAND2015-10662, "Modeling of the Reactor Core Isolation Cooling Response to Beyond Design Basis Operations – Phase 1," (Reference 19) which investigates the manner in which RCIC turbines may respond to increasing water carry-over into the turbine:

> "Preliminary investigations with the RCIC model indicate that liquid water ingestion by the turbine decreases the developed turbine torque; the RCIC speed then slows, and thus the pump flow rate to the RPV decreases. Subsequently, RPV water level decreases due to continued boiling and the liquid fraction flowing to the RCIC decreases, thereby accelerating the RCIC and refilling the RPV. The feedback cycle then repeats itself and/or reaches a quasi-steady equilibrium condition. In other words, the water carry-over is limited by cyclic RCIC performance degradation, and hence the system becomes self-regulating."

Thus, there is significant evidence that credit for RCIC operation with water present in the steam lines is realistic.

Additionally, in the scenarios of interest, operators would then have time to re-establish control of HPCI or RCIC such that the steam lines are not continuously flooded. This specific operator action is not credited, but is implicitly included in the nominal failure probabilities assigned to HPCI and RCIC failing due to the RPV water level initially reaching the main steam lines.

ii. A justification for the nominal failure probability assigned to the failure of the RCIC and HPCI turbines to run after passing liquid, or

Response:

The basic events modeling HPCI or RCIC turbine failures when started with water in the steam lines have a nominal value of 0.05 in the base PRA model. This value was derived based on numerous sets of operator and system manager interviews in which

there was a significant amount of confidence about the ability of the Terry Turbines to successfully perform their function given water in the steam lines. Therefore, a 95% confidence level is assigned based on engineering judgement.

iii. As an alternative to ii., if the RCIC and HPCI turbines were not designed to continue running while and after passing liquid for the conditions that would occur in the accident scenarios they were credited for in the PRA, quantitatively justify that the 10 CFR 50.69 categorization results are not sensitive to this assumption.

Response:

See responses to items i and ii above.

e) Loss of Net Positive Suction Head (NPSH)

Attachment 6 (page 77) of the LAR states that loss of NPSH during certain events (e.g. loss-of-coolant accidents (LOCAs)) leads to the loss of the suppression pool as a cooling inventory source. The LAR explains, however, that this inventory is only needed if the residual heat removal (RHR) system cross-tie fails. The LAR also explains that the licensee assigned a nominal failure probability to provide a slight conservative bias. However, the licensee does not provide information describing the RHR cross-tie or the specific failure mode of the cross-tie. In addition, the licensee does not provide information describing what nominal failure probability value is assigned to this failure mode and why the licensee considered the nominal probability to be conservative. Provide the following:

i. A discussion describing how the RHR is cross-tied to provide a cooling option given loss of the suppression pool as a source of cooling inventory. Identify the specific failure mode that was assigned a nominal failure probability.

Response:

The model reflects logic for the RHR cross-tie and given failure of the cross-tie the model considers loss of NPSH. This is modeled with basic event ZPH-NPSHDXI2.

As part of the Extended Power Uprate (EPU) modifications, PBAPS implemented an RHR cross-tie mitigation strategy for the loss of NPSH scenario.

In LOCAs, the blowdown loads may be slightly higher because of the higher initial power. The GE task analyses confirm that the blowdown loads and SSCs remain acceptable after EPU. This includes the assessment that containment accident pressure is no longer required to ensure NPSH is satisfied for the pumps taking suction from the torus. However, this is contingent upon implementation of the RHR cross-tie and associated HEP to perform the alignment within one hour of a large break LOCA initiator coincident with a containment isolation failure.

One change to the RHR system has been implemented regarding eliminating the need to credit containment accident pressure for design basis LOCA calculations. That is, a split flow alignment of the heat exchangers is employed in response to

LOCA conditions. This has been factored into the risk assessment in the following fashion:

a) The drag valves or orifices between the RHR pump and the RHR heat exchanger are replaced with MOVs with divisional power dependencies.
b) A cross-tie MOV between the A and C RHR pumps (and the B and D RHR pumps) is included to allow for split flow from one RHR pump to discharge to both heat exchangers in the RHR loop.
c) A human error probability (HEP) has been developed to represent the human failure rate associated with aligning suppression pool cooling in a timely fashion given the conditions exist that require the cross-tie to be implemented for success of systems taking suction from the suppression pool. The initial HEP value has been derived at 6.0E-2 for implementation in the PRA model. A longer-term action is also included to reflect the need to align the RHR cross-tie and throttle the flow to maintain NPSH. The HEP value associated with this action is much lower since it only includes the remaining execution steps (i.e., the cognitive contribution to the initial HEP evaluation dominates the failure probability).
d) Logic has been added to the model to include the requirement for success of the cross-tie with flow through both RHR heat exchangers in a loop for the scenarios of interest (i.e., large break LOCA initiator with coincident containment isolation failure).
e) The success criteria for other scenarios (i.e., non-DBA type LOCA scenarios) remain the same in the model.

The enhancements to the PRA model identified in the listing above reflect the plant changes as a result of the EPU modifications and do not introduce new methodology into the PRA model. Therefore, these changes are considered PRA maintenance.

Basic event ZPH-NPSHDXI2 models the failure probability of systems taking suction from the Suppression Pool given the containment overpressure scenario has occurred. The approach to assigning the probability is detailed in response to item ii below.

ii. A justification for the nominal failure probability value, and why this value is valid for this failure mode, or

Response:

Suppression pool inventory may become ineffective as an inventory source due to the loss of NPSH during a large LOCA, or ATWS, or in a transient with failure of suppression pool cooling. In each case, a containment isolation failure could lead to the loss of NPSH which then leads to the loss of the suppression pool for injection. However, NPSH can be maintained through Operator action to utilize the RHR cross-tie (as described above) and throttle injection flow as directed in Transient Response Implementation Plan (TRIP) procedure T-102 (Primary Containment Control) Emergency Operation Procedure (EOP). T-102 directs the operators to control pump flow to remain below the applicable NPSH pump curves provided in T-102. The T-102 Bases note that the NPSH curves are a function of torus level, torus temperature, torus

pressure, and pump flow rates and that the quickest way to ensure operation within the NPSH limit is to reduce pump flow.

Additionally, there is a high likelihood that the operators will be able to throttle the injection flow as necessary in order to maintain NPSH requirements in the postulated scenarios. T-102 directs the Operators to control pump flow to remain below the applicable NPSH pump curves provided in T-102. The T-102 Bases note that the NPSH curves are a function of torus level, torus temperature, torus pressure, and pump flow rates and that the quickest way to ensure operation within the NPSH limit is to reduce pump flow. Due to the primary importance assigned to torus temperature monitoring and the simplicity of the control manipulation to reduce flow, the probability of event ZPH-NPSHDXI2 is set to 0.1.

Given the above, the probability of event ZPH-NPSHDXI2 is set to 0.1. This is considered to be a reasonable probability reflecting the likelihood of occurrence of conditions that may lead to affecting NPSH and Operator response to compensate for this, and the uncertainty in the scenario.

Given the limited scenarios where this assumption is applicable, there is minimal impact on the model results.

iii. As an alternative to ii., justify, quantitatively, that the 10 CFR 50.69 categorization results are not sensitive to this assumption.

Response:

See response to item ii above.

f) Low Intake Pond Level

Attachment 6 (page 78) of the LAR states that in the unlikely event of low intake pond level, risk can be averted by reducing power levels prior to a plant trip or by tripping the circulating water pumps following a plant trip. The LAR states that nominal failure probabilities are assigned to derive the overall likelihood that the precursor events (based on plant experience) proceed to a totally unrecoverable type of event. Provide the following:

i. Description and justification on how the nominal failure probabilities are derived, and the specific failure mode associated with the assigned probabilities.

Response:

The loss of station cooling calculation is described in Appendix D of the PBAPS FPIE PRA Update Initiating Event Notebook. Details of this calculation are provided in the following paragraphs.

"The Peach Bottom site precursor events can be used and estimates of potential recoveries applied. This approach is illustrated by the event tree shown in Figure D-4."

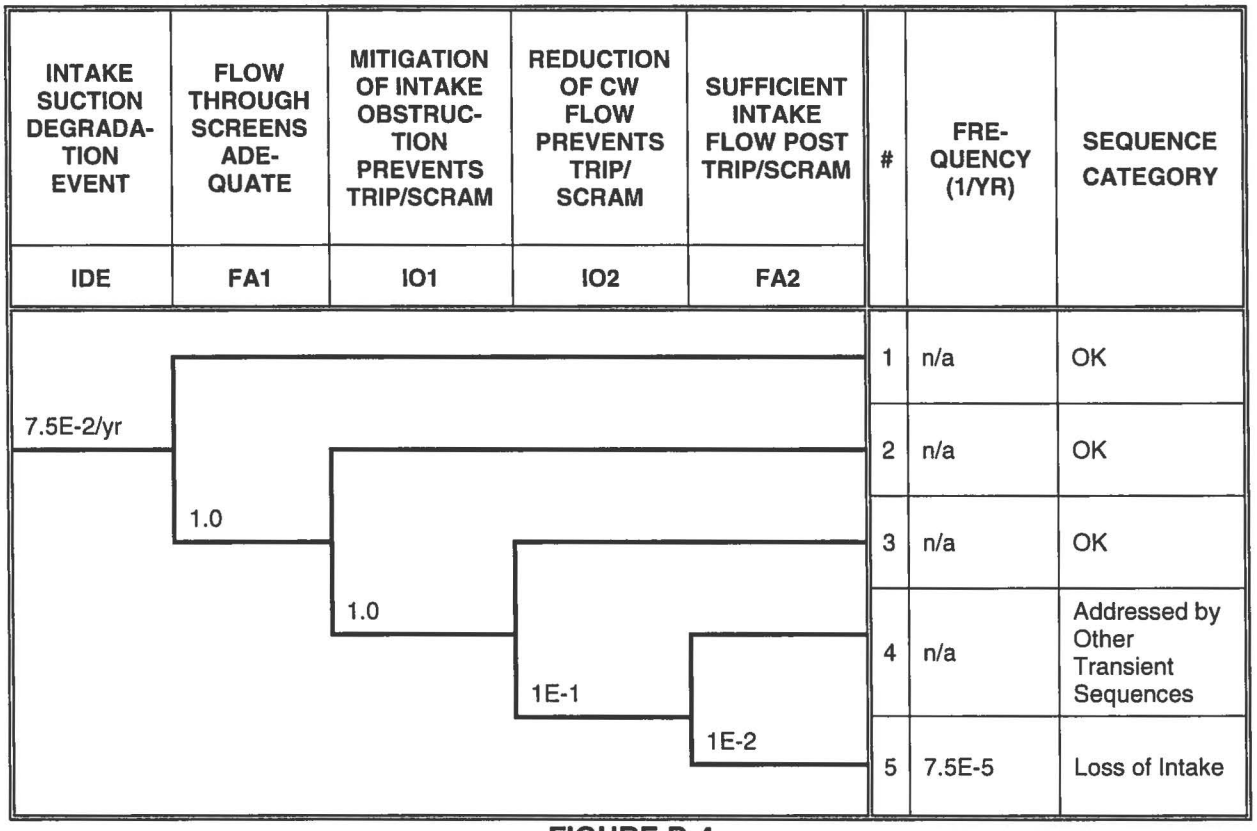


FIGURE D-4
EVENT TREE CALCULATION FOR LOSS OF INTAKE ANNUAL FREQUENCY

"The event tree in Figure D-4 considers the frequency of Peach Bottom precursor events and subsequent credit for operator actions in terminating the event to calculate the annual frequency of experiencing a loss of intake. The nodes in the Figure D-4 event tree are discussed below.

- **IDE** – This event is an estimate of the annual frequency at Peach Bottom of experiencing loss of intake precursors. Three intake flow degradation precursor events have occurred over the operating history of the Peach Bottom site (approximately 40 site years at the time of this calculation; commercial operation began in 1974). Therefore, the frequency for this event is estimated as 3 events / approximately 40 site years = 7.5E-2/yr.
- **FA1** – The down branch of this node is the probability that the loss of intake precursor always results in inadequate flow through the intake screens and low intake level). The probability is assigned a value of 1.0.
- **IO1** – The down branch of this node is an estimate of the probability that plant personnel do not take recovery actions to mitigate the intake obstruction, or that their actions are ineffective or not performed in a timely manner to recover low intake level. A value of 1.0 is assigned to this node.
- **IO2** – The down branch of this node is an estimate of the probability that plant personnel do not take actions to reduce CW suction flow, or that their actions are ineffective or not performed in a timely manner to prevent a

plant trip on low intake level. Peach Bottom experienced a water level drop in the intake structure due to ice blockage in the intake canal in 2004. However, the situation was quickly remedied by lowering plant power and tripping one of the circulating water pumps. The level quickly increased back to normal with only a single circulating water pump in operation. Therefore, reducing power and CW suction flow is a familiar recovery action for loss of intake precursors which can be performed from the Control Room in a matter of minutes. Because this event is familiar at Peach Bottom, a value of 1E-01 is applied to this node.

- **FA2** – Given that plant personnel fail to mitigate the intake challenge in a timely manner (i.e., failure at nodes IO1 and IO2), the plant will scram on loss of condenser vacuum due to loss CW flow or operators will manually shutdown the plant (the more likely scenario). Following the plant trip or shutdown on low intake level the operators are expected to trip the CW pumps (if they have not already done so) and it is very likely that sufficient intake flow and level will be restored. However, a conditional failure probability of 1E-2 is assigned to this node to represent the likelihood that the plant shutdown and trip of the CW pumps does not result in recovery of intake level."

As can be seen from Figure D-4 above, the frequency of a complete loss of station cooling via the intake structure at the Peach Bottom site may be estimated at 7.5E-5/yr when considering the precursor events; including the experience from the actual 2004 event.

It is important to note that should the site experience such an intake degradation event, there is another separate and redundant mechanism of supplying emergency cooling to safety related equipment via closed loop cooling to safety related equipment. The PRA models this feature.

ii. Confirmation and description of the two operator actions cited above are proceduralized steps required to be taken when the intake pond level is low enough to fail this inventory source, or

Response:

Operator response to lowering river/pond events are included in Special Event (SE) procedure SE-3 (Loss of Conowingo Pond) as well as Abnormal Operating (AO) procedure AO 29.2.

SE-3 is entered when there is "any indication that the continued use of the Conowingo Pond as a heat sink is threatened."

SE-3 directs that if there is an unscheduled drop in river OR intake canal level to 104 ft., Operators are directed to shut down both reactors and proceed to MODE 4 using General Plant (GP) procedure GP-3 (Normal Plant Shutdown). The Operators would lower power levels as part of this procedure.

If there was an unscheduled drop in river or intake canal level to 98.5 ft, Operators are

directed to SCRAM both reactors and enter TRIP procedures T-100 or T-101. Once the reactors are both shutdown, the Operators monitor the circulating water pumps per the procedural guidance and are directed to secure them prior to any cavitation issues that could occur starting at 92.5 ft.

SE-3 directs Operators to enter AO 29.2 (Discharge Canal to Intake Pond Cross-Tie Gate Operation, Frazil Ice Mitigation and Icing Condition Operations) if intake canal level cannot be maintained. AO 29.2 also provides Operators with the means to reduce circulating water pump operation in support of the Discharge Canal to Intake Pond cross-tie gate operation.

iii. As an alternative to ii., justify, quantitatively, that the 10 CFR 50.69 categorization results are not sensitive to this assumption.

Response:

See response to item ii above.

g) HEP Dependency Analysis

Attachment 6 (page 80) of the LAR identifies dependent Human Error Probability (HEP) values as a source of uncertainty and cites the sensitivity study on HEPs performed in accordance with Section 5 of NEI 00-04 as the disposition. Clarify how the sensitivity studies performed for HEPs will address HEP dependency uncertainty.

Response:

As directed by NEI 00-04, internal events human error basic events are increased to their 95th percentile and also decreased to their 5th percentile values as part of the required 50.69 PRA categorization sensitivity cases. As a part of these sensitivities, the dependent HEPs are also set to their 95th and 5th values for the respective sensitivity cases.

h) Pipe Rupture due to Water Hammer

Attachment 6 (page 82) of the LAR states that there is uncertainty associated with the success of RHR when the reactor is in suppression pool cooling mode because of the potential for a pipe rupture following a water hammer. The LAR states that the water hammer events and values utilized provide a reasonable approach. However the licensee does not describe nor justify the approach. Provide the following:

i. A description of and a basis for the approach used to model the potential for pipe rupture in the RHR system following a water hammer while in suppression pool cooling mode or,

Response:

PRA Modeled Calculation

The water hammer related failure probability calculations used in the PRA model are

based on:

- A comparison of water hammer loads relative to a 1983 event
- Industry data assessment

Based on this analysis, if a water hammer occurs in a line, the possibilities are as follows:

1. No component damage will occur (no pressure boundary failure or flow blockage) = P_{ND}
2. Line blockage will occur (with no pressure boundary failure) = P_B
3. A small leak (<1200gpm) will occur = P_{SL}
4. A large break (>1200gpm) will occur = P_{LL}

The following values are assigned to these possible outcomes: P_{SL}=0.09, P_{LL}=0.01, and P_B=0.01, leaving the probability of no damage to be P_{ND} =0.89. The following basic events and values are used to implement this in the Peach Bottom PRA model: DPH-LEAKWH=0.09, DPH-RUPTWH=0.01, and DPH-PLUGWH=0.01.

Conclusion

The PRA modeling of the water hammer scenario may represent a slight conservative bias but is considered to provide a reasonable best-estimate approach for this event given the minimal available industry evidence.

ii. Justify, quantitatively, that the 10 CFR 50.69 categorization results are not sensitive to this treatment.

Response:

The model was run to evaluate how sensitive the PRA results are to adjustments made to the water hammer failure probabilities (leakage, rupture, and blockage). These failure probabilities were increased by a factor 5 and the results were analyzed.

The sensitivity case failure probabilities for the water hammer events are as follows:

P_{SL}	=	0.45	(basic event DPH-LEAKWH)
P_{LL}	=	0.05	(basic event DPH-RUPTWH)
P_B	=	0.05	(basic event DPH-PLUGWH)

There were no new basic events identified in the sensitivity case results that exceeded the FV or RAW HSS threshold and that were not already HSS from the base case model quantifications. Therefore, these sensitivity case results demonstrate that the PRA model is not sensitive to the currently modeled water hammer failure probabilities.

RAI 09 – Overall Categorization Process

Section 50.69(b)(2)(i) requires that a licensee's application contain a description of the process for SSC categorization. The discussion in Section 3.1.1 of the LAR does not provide enough information for the NRC staff to clearly understand the sequence of evaluations in the categorization process, what information the licensee will develop and use, and what guidance the licensee will follow for the Integrated Decision-making Panel (IDP) decision-making process 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 response to this RAI. Provide the following:

a. Summarize, in the order they will be performed, the sequence of elements or steps that the licensee will follow for each system that will be categorized. A flow chart, such as that provided in the September 6, 2017 public meeting with Nuclear Energy Institute (NEI) (ADAMS Accession No. ML17249A072) regarding 10 CFR 50.69 License Amendment Requests may be provided instead of a description. The public meeting summary is available at ADAMS Accession No. ML17265A020. The steps should include:

    i. The input from all PRA evaluations such as use of the results from the IEPRA, internal flooding, and FPRAs;

    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 description to item (a) above, please clarify the difference between "preliminary high-safety significance (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. Confirm that the proposed approach is consistent with the guidance in NEI 00-04, as endorsed by RG 1.201.

c. In description to item (a) above, please 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.

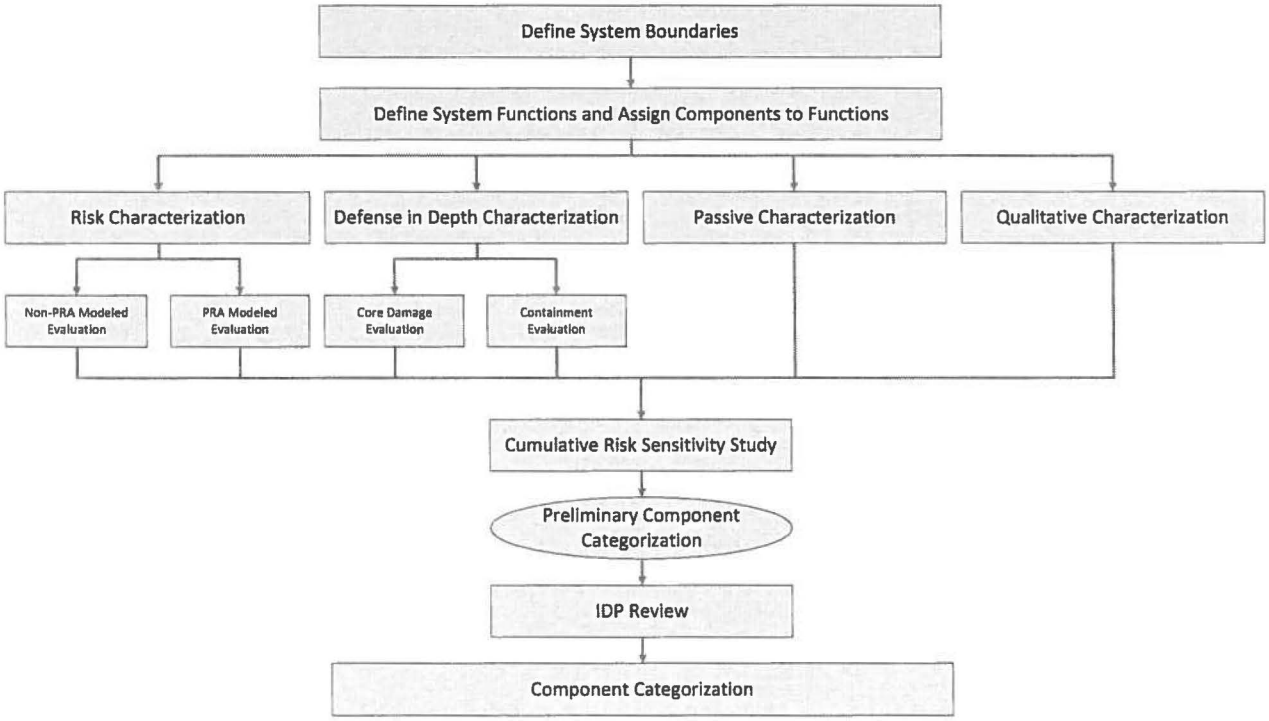
Responses to a), b), and c):

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 9-1: Categorization Process Overview



Categorization of SSCs will be completed per the NEI 00-04 process, as endorsed by RG 1.201, which includes the determination of safety significance through the various elements identified above. The results of these elements are used as inputs to arrive at a preliminary component categorization (i.e., 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 below 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 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 of preliminary safety significance. 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, and 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 to d):

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 by NEI 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 to e):

Please see the response to Questions 09a, 09b, and 09c. 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 10 – SSCs Categorization based on External Hazards

Sections 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. The information provided in the LAR is not sufficient for the NRC to determine if the licensee's proposed categorization process is consistent with the guidance in NEI 00-04. The specific hazards and requested information are as follows:

a) "Other" External Hazards

Section 3.2.4 of the LAR states that the categorization process will use screening results from the Individual Plant Evaluation of External Events (IPEEE), performed in response to GL 88-20, "Individual Plant Examination for Severe Accident Vulnerabilities," for evaluation of safety significance related to the other external hazards and that all 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 in NEI 00-04. Provide the following:

   i. 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.

ii. Identify which hazards will have all SSCs credited and 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.

As part of the external hazard screening noted in the response to item i above of this question, an evaluation was performed to determine if there are components that participate in screened scenarios and whose failure would result in an unscreened scenario. Refer to the response to Part c) of this question for SSCs credited in the extreme wind or tornado hazard screening.

iii. A description of, and justification for, any additional method(s) different from (i) or (ii) above that the licensee will use to evaluate individual SSCs against "other" external hazards and identify the hazards that the licensee will evaluate with these methods.

Response:

There are no additional method(s) different from items i or ii that will be used to evaluate individual SSCs against external hazards.

iv. Confirm that all hazards not included in the categorization process in response to (i), (ii), or (iii) 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 items i, ii, or iii above are considered insignificant for every SSC and, therefore, will not be considered during the categorization process.

b) External Flooding

Figure 5-6 of NEI 00-04 shows that if a component is included in a screened scenario, then in order for that component to be considered a candidate LSS, the licensee has to show that if the component was removed, the screened scenario would not become unscreened. Attachment 4 of the LAR indicates that the licensee screens external

flooding hazards out of the 10 CFR 50.69 process. The LAR states that flooding from rivers and streams (precipitation based), and dam failure are bounded by the current licensing basis (CLB) and do not pose a challenge to the plant, and that flooding from local intense precipitation is not bounded by the current licensing basis. However, the licensee does not provide any information describing SSCs, if any, that are included in the screening. Provide the following:

i. A discussion identifying SSCs, if any, that are credited in the screening of external flooding, including passive and/or active components

Response:

The NRC staff issued its assessment of the PBAPS flooding focused evaluation in a letter dated November 6, 2017 (Reference 20), "Peach Bottom Atomic Power Station, Units 2 and 3 - Staff Assessment of Flooding Focused Evaluation (CAC Nos. MG0092 and MG0093)." As described in the assessment letter, there are no SSCs credited in the screening of external flooding. Either available physical margin exists or, where water ingress is expected, all external flood mechanisms resulted in Water Surface Elevations (WSEs) below the design basis protection level of the plant. The following provides additional detail from the flooding focused evaluation supporting this conclusion.

As stated in Reference 20, storm surge, seiche, ice-induced flooding, and local intense precipitation (LIP) flood-causing mechanisms were not bounded by the plant's current design basis and therefore additional assessments of these flood hazard mechanisms were performed. For storm surge, seiche, and ice-induced flooding, the station concluded and the NRC agreed that adequate margin exists and that the natural topography around the site provides protection such that the events will not impact key SSCs.

For Local Intense Precipitation (LIP) flood-causing mechanisms, the focused evaluation discussed areas in which physical margin is available and those in which water ingress is expected. As discussed in the PBAPS walkdown report (Reference 21) and in the NRC staff's letter (Reference 20), the Emergency Cooling Tower (ECT), Emergency Diesel Generator Building (EDGB), and the Emergency Pump Structure have available physical margin above the respective water ingress levels. Further, although water is expected to ingress into the reactor building, PBAPS performed a technical evaluation to determine the in-leakage and concluded that the volume was such that equipment operation would not be adversely affected.

ii. A discussion explaining how the licensee will apply the guidance in Figure 5-6 of NEI 00-04 to any SSCs that are credited for screening of external flooding.

Response:

As discussed in item i above, there are no SSCs credited for screening of external flooding.

c) High Winds

Figure 5-6 of NEI 00-04 shows that if a component is included in a screened scenario, then in order for that component to be considered a candidate LSS, the licensee has to show that if the component was removed, the screened scenario would not become unscreened. Attachment 4 of the LAR states that the licensee screens out the extreme wind or tornado hazard on the basis that the frequency of damage to the exposed components is estimated to be less than 1E-6/year. However, the licensee does not provide any information describing the SSCs, if any, that are included in the screening. Provide the following:

i. A discussion identifying the SSCs, if any, that are credited in the screening of extreme wind and tornados, including passive and/or active components.

ii. A discussion explaining how the licensee will apply the guidance in Figure 5-6 of NEI 00-04 to any SSCs that are credited for screening of extreme wind and tornados.

iii. Explain how the discussion in items (i) and (ii) above would be impacted by the current effort to assess tornado missile protection hazard in response to RIS 2015-06 "Tornado Missile Protection," (ADAMS Accession No. ML15020A419).

Responses to i, ii, and iii above:

In support of the station's continuing efforts to assess tornado missile protection hazards in response to RIS 2015-06, Exelon is requesting additional time to develop an appropriate response to the issues described in the above question. Exelon will provide a supplemental response to this RAI question with a response date currently expected to be no later than June 6, 2018.

RAI 11 – Shutdown Risk

Section 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. Section 3.2.5 of the LAR states that the 10 CFR 50.69 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 and shutdown conditions. However, the LAR does not cite the other categorization criteria listed in NEI 00-04 Section 5.5 pertaining to shutdown events (e.g., DID attributes, 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 Peach Bottom, NEI 00-04, as endorsed by RG 1.201, allows the use of a process based on the NUMARC 91-06 program. Peach Bottom's categorization process will follow 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 12 – Qualitative Assessments

When classifying SSCs according to risk significance, 10 CFR 50.69(c)(1)(ii) states that a licensee must determine SSC functional importance using an integrated, systematic process for addressing initiating events (internal and external), SSCs, and plant operating modes, including those not modeled in the plant-specific PRA. The functions to be identified and considered include design bases functions and functions credited for mitigation and prevention of severe accidents. The systematic process outlined in NEI 00-04 includes qualitative assessment questions to assist the IDP in determining whether functions/SSCs identified as candidate LSS that are implicitly depended upon to maintain safe shutdown capability, prevention of core damage and maintenance of containment integrity. The staff notes that, unlike the DID assessment outlined in Section 6 of NEI 00-04, the qualitative questions for the IDP in Section 9 are not limited to design-basis events.

To address the requirements in 10 CFR 50.69(c)(1)(ii), describe and justify, with an example, such as FLEX functions, how the severe accident functions will be considered during the categorization process, including discussion on how each of the seven qualitative questions in Section 9 of NEI 00-04 would be applied to those functions.

Response:

Severe accident prevention functions/components are evaluated with the categorization process as outlined in the RAI #9 response. To illustrate this process for a severe accident prevention function, the categorization process for the FLEX RPV injection flow path function (10-FLEX) of the Peach Bottom RHR system is discussed below.

Risk (PRA Modeled): Severe accident prevention functions/components are evaluated using PRA model results in the same way as all other system functions in accordance with Table 1 (see response to RAI 9).

Function 10-FLEX is a non-modeled function for both the FPIE and Fire PRA models.

Risk (Non-modeled): Severe accident prevention functions/components are evaluated using non-modeled methods in the same way as all other system functions/components in accordance with Table 1 (see response to RAI 9).

The seismic hazard is evaluated solely at a component level and, therefore, has no direct effect on the categorization of Function 10-FLEX. Function 10-FLEX is determined not to be the primary or first alternative method of achieving any Shutdown Key Safety Functions (i.e., does not support a primary shutdown safety system) nor are FLEX components able to initiate a Shutdown Event. Therefore, Function 10-FLEX is categorized as LSS with respect to Shutdown Risk.

Defense-in-Depth:

Core Damage - Severe accident prevention functions/components are used to address Beyond Design Basis Events (BDBEs). These are not "internally initiated design basis events" [NEI 00-04, Page 49]; therefore, the core damage defense-in-depth assessment is not applicable.

Containment - Severe accident prevention functions/components are evaluated using the containment defense-in-depth questions in the same way as other system functions/components in accordance with Table 1 (see response to RAI 9).

The core damage defense-in-depth assessment is not applicable for Function 10-FLEX. Function 10-FLEX is determined not to be an associated function for any component that required a "Yes" response to a containment defense-in-depth question. Therefore, Function 10-FLEX is categorized as LSS with respect to defense-in-depth.

Qualitative Criteria: Severe accident prevention functions/components (BDBE functions/SSCs) are evaluated using the qualitative criteria (7 Considerations) in the same way as all other system functions in accordance with Table 1 (see response to RAI 9). Refer to NEI 00-04 pg. 65-66 for the wording of each consideration. The considerations are preliminarily answered by the 50.69 categorization team as described below; however, it should be noted that ultimately these considerations lie within the purview of the Integrated Decision-Making Panel members. The final interpretation of these considerations with respect to severe accident prevention functions is also the responsibility of the Integrated Decision-Making Panel.

Consideration #1: SSCs mapped to severe accident prevention functions are typically isolated from normal plant operating equipment and would not cause an initiating event. However, in the unlikely case that this consideration is answered false for the BDBE function/SSC, then the BDBE function would be categorized as HSS due to this consideration.

Consideration #2: Components in severe accident prevention functions are not typically part of the RCS pressure boundary; however, if they are such that their failure could result in leakage beyond normal makeup capacity, then the function would be categorized as HSS due to this consideration.

Consideration #3: Components in severe accident prevention functions are typically isolated such that their failure would not affect the remaining defense-in-depth available to achieve basic safety functions (i.e. the frontline systems credited to maintain reactivity control, core cooling, heat sink, and RCS inventory). If a severe accident prevention function fails in such a way that it challenges a frontline system, then the function would be categorized as HSS due to this consideration.

Consideration #4: "Emergency/Abnormal Operating Procedures or similar guidance" used to mitigate an accident or transient are considered to be the base set of Emergency Operating Procedures. These procedures contain the essential guidance required for successful performance of operator actions required for DBE accident/transient response. FLEX procedures and Severe Accident Mitigating Guidelines are not part of the base set of EOPs and therefore are not applicable for this consideration. However, if a BDBE function is used in such a way that it is the sole means of progressing through the procedures, then the function would be categorized as HSS due to this consideration.

Consideration #5: "Emergency/Abnormal Operating Procedures or similar guidance" are considered to be the base set of Emergency Operating Procedures and also Emergency Planning procedures. These procedures contain the essential guidance required for achievement of actions related to assuring long term containment integrity, monitoring of post-accident conditions, and/or emergency planning activities. FLEX procedures and Severe Accident Mitigating Guidelines are not used in the base set of Emergency Operating Procedures or Emergency Planning procedures; however, if a BDBE function is used in such a way that it is the sole means of progressing through the procedures, then the function would be categorized as HSS due to this consideration.

Consideration #6: Severe accident prevention functions are typically not used to reach or maintain safe shutdown conditions. The design basis frontline systems are generally available and used for this purpose; therefore, severe accident prevention functions would typically not be considered essential for reaching or maintaining safe shutdown conditions. If a severe accident prevention function was to be essential for this purpose, then the function would be categorized as HSS due to this consideration.

Consideration #7: Components in severe accident prevention functions are not typically part of the three classical fission product barriers (fuel cladding, RCS pressure boundary, and primary containment); however, if they are identified as part of one of these boundaries such that this consideration is false, then the function would be categorized as HSS due to this consideration.

The considerations are addressed for Function 10-FLEX in the typical manner described above;

Function 10-FLEX is categorized as LSS due to the Qualitative Criteria.

Passive: The passive categorization process evaluates segments/components. This process impacts the categorization of components within severe accident prevention functions; however, there is no direct effect on the function due to the passive categorization process.

The passive categorization has no direct effect on the categorization of Function 10-FLEX.

RAI 13 – Interfacing SSCs

On January 31, 2018, the NRC staff observed the deliberations of the IDP for the RHR System. The staff observed that the RHR categorization process appeared to categorize RHR functions/SSCs that interfaced with other systems that have not yet been categorized, such as the Automatic Depressurization System (ADS) and Core Spray System. The guidance in Section 4 of NEI 00-04 states that a candidate LSS SSC that supports an interfacing system should remain uncategorized until all interfacing systems are categorized.

a. Confirm that the guidance in NEI 00-04 will be followed and that any functions/SSCs that serve as an interface between two or more systems will not be categorized until the categorization for all of the systems that they support is completed, or otherwise justify your proposed approach.

Response to 13.a:

The guidance in NEI 00-04 that any functions/SSCs that serve as an interface between two or more systems will not be categorized until the categorization for all the systems they support is completed will be followed.

b. Confirm that the item in a. above will be captured in the 50.69 categorization procedures.

Response to 13.b:

The guidance referred to in item #13.a above is captured in the 50.69 categorization procedures.

RAI 14 – F&O Closeout Process

- 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 LAR. On January 31 and February 1, 2018, the NRC staff performed an audit of the 50.69 categorization process, including a review of the revised F&O Closure Review Report and the revised PRA Technical Adequacy Evaluation Report. The staff noticed several inconsistencies between the information provided in the revised F&O closure report and the revised PRA technical adequacy evaluation, as compared to the information provided in the LAR and LAR supplement. Examples of the discrepancies include1) Peer review F&Os not included in the LAR that were not addressed in the F&O closure report, and 2) open F&Os listed in the F&O closure report that were not provided in the LAR.

Provide the following:

a. An explanation of how the F&Os were selected for the F&O closure process and how/why that differs from those provided in the LAR.

Response:

All finding F&Os that were considered internally addressed as of the time of the F&O Closure Review were selected for the F&O Closure Review process.

Any finding F&Os that were self-identified as open at the time of the F&O Closure Review were not provided to the F&O Closure Review team for review.

b. Because the scope of the 10 CFR 50.69 categorization process is so broad, all open F&Os may potentially be relevant to this LAR and could have an impact on the 10 CFR 50.69 categorization results. Provide any outstanding F&Os that have not been closed by the F&O closure review and have not been provided in the LAR. For each F&O include:

   i. The complete F&O description, the peer review team recommendation, the licensee's resolution, or disposition for the application, and the F&O closure team assessment, if applicable.

   Response:

   During the review, Exelon identified three outstanding F&Os that have not been closed by the F&O closure review and were not provided in the LAR. These are discussed in the following table (Table 2).

TABLE 2
OPEN AND PARTIALLY RESOLVED PEER REVIEW FINDINGS NOT PRESENTED TO F&O CLOSURE REVIEW TEAM OR PROVIDED IN LAR SUBMITTAL

Finding Number	Supporting Requirement(s)	Capability Category (CC)	Finding Description	Finding Basis	Proposed Finding Resolution	Maintenance or Upgrade	Disposition for 50.69
2011-6-11	DA-C11	Not Met	The data from the Maintenance Rule is used directly without checking to see if it includes only those maintenance or test activities that could leave the component, train, or system unable to perform its function when demanded as required by the SR.	The F&O needs to be addressed to meet the SR.	Study the Maintenance Rule data and estimate the specific values required by the SR.	Maintenance Basis: This change is documentation related and does not require the use of any methods not already peer reviewed for the PBAPS PRA. Therefore, this change is considered maintenance.	NOT PRESENTED TO CLOSURE REVIEW TEAM PARTIALLY RESOLVED – OPEN DOCUMENTATION Final Disposition: As resolution of this finding is a documentation enhancement there will be no impact on the 50.69 categorization process. Exelon Procedure No. WC-AA-101 provides guidelines for how to track component, train, or system unavailability and states the following for MRULE monitoring: "SSCs out of service for testing are considered unavailable, unless the function can be promptly restored either by an operator in the control room or by a dedicated operator stationed locally for that purpose. Restoration actions must be contained in a written procedure, must be uncomplicated (a single action or a few simple actions), and must not require diagnosis or

TABLE 2
OPEN AND PARTIALLY RESOLVED PEER REVIEW FINDINGS NOT PRESENTED TO F&O CLOSURE REVIEW TEAM OR PROVIDED IN LAR SUBMITTAL

Finding Number	Supporting Requirement(s)	Capability Category (CC)	Finding Description	Finding Basis	Proposed Finding Resolution	Maintenance or Upgrade	Disposition for 50.69
							repair. Credit for a dedicated local operator can be taken only if (s)he is positioned at the proper location throughout the duration of the test for the purpose of restoration of the train should a valid demand occur. The intent of this paragraph is to allow licensees to take credit for restoration actions that are virtually certain to be successful (i.e., probability nearly equal to 1) during accident conditions." Therefore, using raw data taken directly from the Maintenance Rule provides very realistic testing and maintenance failure probabilities for the PRA model as any testing or maintenance that would not make the SSC unavailable would not be tracked in MRULE.
2011-6-14	IFSN-A17	Met CC I/II/III	Plant walk down was conducted to identify flood sources. No specific walkdown was conducted to	Needed to improve the quality for the flood PRA.	While doing a future walkdown, prepare a walkdown sheet and have the analyst fill in all the needed	Maintenance Basis: This change is documentation related and does not require the use of any methods not	CLOSURE REVIEW TEAM ASSESSMENT: OPEN Review Team Assessment: During the on-site review, the following three documents were provided: • PB-PRA-012A; Internal Flood Walkdown Notebook

TABLE 2
OPEN AND PARTIALLY RESOLVED PEER REVIEW FINDINGS NOT PRESENTED TO F&O CLOSURE REVIEW TEAM OR PROVIDED IN LAR SUBMITTAL

Finding Number	Supporting Requirement(s)	Capability Category (CC)	Finding Description	Finding Basis	Proposed Finding Resolution	Maintenance or Upgrade	Disposition for 50.69
			identify the SSCs in the flood areas or the pathways. These were identified through drawings, and verified by mini-walkdowns at the discretion of the PRA analysts. The walkdown documentation is very sketchy. A lot more information needs to be collected during walkdown to help flood scenario development. The location of drains, curbs, doors, sills need to		information regarding flood sources, SSCs, flood propagation and mitigation.	already peer reviewed for the PBAPS PRA. Therefore, this change is considered maintenance.	• PBAPS PRA Events – Hand written walkdown notes with information on flood "targets" in the plant areas • PBAPS Water Sources – Hand written walkdown notes with information on flood sources in the plant areas The first document is an unsigned MSWord file. The latter two are PDF files that are intended to serve as Appendix A to the first document. Review Team Recommendation: Confirm that this document exists in final, signed-off form with the Appendix included. Final Disposition: The recommendation of the F&O Technical Review team was to sign-off the Internal Flood Walkdown Notebook. This Notebook includes Appendices A and B which document the flood source and target notes, respectively. The Internal Flood Walkdown Notebook has subsequently been signed off,

TABLE 2
OPEN AND PARTIALLY RESOLVED PEER REVIEW FINDINGS NOT PRESENTED TO F&O CLOSURE REVIEW TEAM OR PROVIDED IN LAR SUBMITTAL

Finding Number	Supporting Requirement(s)	Capability Category (CC)	Finding Description	Finding Basis	Proposed Finding Resolution	Maintenance or Upgrade	Disposition for 50.69
			be identified. The paths through stairwells need to be identified. The flood pathways developed in the flood scenarios need to be verified by walkdown.				and, therefore this finding is no longer relevant.
2011-6-15	IFSN-A8	Met CC I	Inter-area propagation has been addressed in the scenario development. However, flow path via drain lines, and areas connected via backflow through drain lines involving failed check valves, pipe and cable	Need to address this to meet CC II requirements.	In the scenario development, include consideration of the flow paths identified.	Maintenance Basis: This change does not require the use of any methods not already peer reviewed for the PBAPS PRA and will not result in any new accident sequences. Therefore, this change does not constitute a model upgrade.	CLOSURE REVIEW TEAM ASSESSMENT: OPEN Review Team Assessment: While a lot of work has been done, and they have identified propagation between FL39 and FL40. However, it is not properly included in the model. Review Team Recommendation: Include the propagation scenario in the model or otherwise disposition it on a technical basis. Final Disposition: The recommendation of the F&O

TABLE 2
OPEN AND PARTIALLY RESOLVED PEER REVIEW FINDINGS NOT PRESENTED TO F&O CLOSURE REVIEW TEAM OR PROVIDED IN LAR SUBMITTAL

Finding Number	Supporting Requirement(s)	Capability Category (CC)	Finding Description	Finding Basis	Proposed Finding Resolution	Maintenance or Upgrade	Disposition for 50.69
			penetrations (including cable trays) do not appear to be addressed.				Technical Review team was to evaluate the propagation between flood scenarios FL39 and FL40. The PRA models (PB214A5 and PB214A5F0, similar for Unit 3) developed in PB-ASM-11 were updated to include 8 new flooding scenarios between both Unit models which account for this propagation. Therefore, there is no impact to the application as this F&O has subsequently been addressed and the model used for 50.69 categorization will reflect this change.

ii. The licensee's documented justification supporting the classification of each F&O finding resolution for open F&Os as either a PRA "upgrade" or PRA "maintenance update", as defined in the ASME/ANS RA-Sa-2009 PRA Standard endorsed by RG 1.200, Revision 2.

Response:

See response to item i above.

RAI 15 – Passive Component Categorization

LAR Section 3.1.2 stated that for the categorization of passive components and the passive function of active components, the licensee will use 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 conditional core damage probability (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 No. 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 Peach Bottom for 10 CFR 50.69 SSC categorization.

References:

1. Letter from James Barstow (Exelon Generation Company, LLC) to U.S. Nuclear Regulatory Commission – "Application to Adopt 10 CFR 50.69, 'Risk-informed categorization and treatment of structures, systems, and components for nuclear power plants,'" dated August 30, 2017 (ML17243A014)

2. Letter from Richard B. Ennis (U.S. Nuclear Regulatory Commission) to Bryan C. Hanson, Exelon Generation Company, LLC – "Supplemental Information Needed for Acceptance of Requested Licensing Action Re: Adoption of title 10 of the Code of Federal Regulations Section 50.69," dated October 10, 2017 (ML17272B016)

3. Letter from James Barstow (Exelon Generation Company, LLC) to U.S. Nuclear Regulatory Commission – "Supplement to Application to Adopt 10 CFR 50.69, Risk-Informed Categorization and Treatment of Structures, Systems, and Components for Nuclear Power Reactors," dated October 24, 2017 (ML17297B521)

4. Electronic mail message from Jennifer Tobin (U.S. Nuclear Regulatory Commission) to David Helker, Exelon Generation Company, LLC - "Draft 50.69 Request for Additional Information (RAIs) - Peach Bottom," dated March 21, 2018

5. Electronic mail message from Jennifer Tobin (U.S. Nuclear Regulatory Commission) to David Helker, Exelon Generation Company, LLC – "Peach Bottom Units 2 and 3 - Request for Additional Information - Adopt 50.69 License Amendment," dated April 6, 2018

6. Peach Bottom Atomic Power Station PRA Peer Review Report Using ASME/ANS PRA Standard Requirements, May 2011.

7. Peach Bottom Atomic Power Station Fire PRA Peer Review Report Using ASME/ANS PRA Standard Requirements, April 2013.

8. JENSEN HUGHES Report 032299-RPT-001 Revision 1, "Peach Bottom Atomic Power Station PRA Finding Level Fact and Observation Technical Review & Focused-Scope Peer Review," October 2017.

9. NUREG/CR-6931, Volume 3, "Cable Response to Live Fire (CAROLFIRE) Volume 3: Thermally-Induced Electrical Failure (THIEF) Model," (ADAMS Accession No. ML081190261)

10. NUREG/CR-7010, Volume 1, "Cable Heat Release, Ignition, and Spread in Tray Installations During Fire (CHRISTIFIRE)," (ADAMS Accession No. ML12213A056)

11. NUREG/CR-6850, "Electric Power Research Institute (EPRI)/NRC-RES Fire PRA Methodology for Nuclear Power Facilities," (ADAMS Accession Nos. ML052580075, ML052580118, and ML103090242)

12. ASME/ANS RA-S-2009, "Addenda to ASME/ANS RA-S-2008, Standard for Level1/Large Early Release Frequency Probabilistic Risk Assessment for Nuclear Power Plant Applications," American Society of Mechanical Engineers, 2009

13. NUREG-7010, "Cable Heat Release, Ignition, and Spread in Tray Installations During Fire, Volume 1 Horizontal Trays" (ADAMS Accession No. ML12213A056)

14. NUREG-1921/EPRI 1023001, "EPRI/NRC-RES Fire Human Reliability Analysis Guidelines" (ADAMS Accession No. ML12216A104)

15. NEI 00-01, Revision 3, "Guidance for Post Fire Safe Shutdown Circuit Analysis," Nuclear Energy Institute, October 2011 (ADAMS Accession No. ML112910147)

16. NUREG-1824/EPRI 1011999, "Verification and Validation of Selected Fire Models for Nuclear Power Plant Applications Volume 1" (ADAMS Accession No. ML071650546)

17. EPRI 1020236, "MAAP4 Applications Guidance," Electric Power Research Institute, July 2010

18. NUREG/CR-6928, "Industry Average Performance for Components and Initiating Events at U.S. Commercial Nuclear Power Plants," (ADAMS Accession No. ML070650650) Component Reliability 2010 update.

19. SAND2015-10662, ""Modeling of the Reactor Core Isolation Cooling Response to Beyond Design Basis Operations – Phase 1," Sandia National Laboratories, November 2015

20. "Peach Bottom Atomic Power Station, Units 2 and 3 - Staff Assessment of Flooding Focused Evaluation (CAC Nos. MG0092 and MG0093)," November 6, 2017 (ADAMS Accession No. ML 17292B763)

21. RS-12-174, "Exelon Generation Company, LLC's 180-day Response to NRC Request for Information Pursuant to 10 CFR 50.54(f) Regarding the Flooding Aspects of Recommendation 2.3 of the Near-Term Task Force Review of Insights from the Fukushima Dai-ichi Accident," November 19, 2012 (ADAMS Accession No. ML123250714)

22. NUREG-1805, Supplement 1, "Fire Dynamics Tools (FDTs) Quantitative Fire Hazard Analysis Methods for the U.S. Nuclear Regulatory Commission Fire Protection Inspection Program," U.S. NRC-NRR, July 2013

23. U.S. Nuclear Regulatory Commission, "CCF Parameter Estimations, 2012 Update," Idaho National Laboratories, November 2013

24. Fire-Induced Vulnerability Evaluation (FIVE) User's Guide: Revision 2. EPRI, Palo Alto, CA: 2014. 3002000830

25. Terry Steam Turbine Division, Terry Corporation, Bulletin No. S-213, Reprint from Power Magazine, October 1973 article: "Nuclear Plants Demand Rugged Auxiliaries – A series of Water Slug Tests Demonstrated the Ability of Steam Turbines to Withstand Shock of Rapid Startup Without Draining Steam Line Beforehand"

26. V. Bochnak, Terry Steam Turbine Company, to L. Holthausen, General Electric Co., "Water Startup and Slug Capability Guarantee," November 14, 1969; and R.E. Fellenz, Terry Steam Turbine Company, to B. Langlais, General Electric Co., "Water Start-up and Slug Capability Guarantee", November 19, 1979

27. V. Bochnak, Terry Steam Turbine Company, to Lee Holthausen, General Electric Co., "RCIC Turbines"; and V. Bochnak, Terry Steam Turbine Company, to Lee Holthausen, General Electric Co., "RCIC Turbines Water Start-Up and Slug Capability Guarantee," June 18, 1973

ATTACHMENT 2

License Amendment Request

Peach Bottom Atomic Power Station, Units 2 and 3
Docket Nos. 50-277 and 50-278

Response to Request for Additional Information
Application to Adopt 10 CFR 50.69, "Risk-informed categorization and treatment of structures, systems, and components for nuclear power plants"

Peach Bottom 50.69 PRA Implementation Items

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 issues identified below will be addressed and any associated changes will be made, focused-scope peer reviews will be performed on changes that are Probabilistic Risk Assessment (PRA) upgrades as defined in the PRA Standard ASME/ANS RA-Sa-2009, as endorsed by Regulatory Guide (RG) 1.200, Revision 2, "An Approach for Determining the Technical Adequacy of Probabilistic Risk Assessment Results for Risk-Informed Activities," and any findings will be resolved and reflected in the PRA of record prior to implementation of the 10 CFR 50.69 categorization process.

Peach Bottom 50.69 PRA Implementation Items	
Item #	Description
1	The HRA pre-initiators in the internal events PRA model will be updated to meet Capability Category II of the ASME/ANS RA-Sa-2009 as endorsed by RG 1.200, Revision 2. A focused-scope peer review will be conducted of the pre-initiator analysis, and any resulting F&Os will be resolved, as indicated in response to RAI 03.a contained in Exelon letter dated May 7, 2018.
2	The transient floor area ratios (FARs) in the fire PRA will be adjusted to consider the treatment of obstructed floor space to provide a more accurate distribution of transient ignition frequency, as indicated in response to RAI 03.b contained in Exelon letter dated May 7, 2018.
3	Risk significant fire PRA scenarios for ignition sources capable of being modeled with a two-point fire modeling approach will be updated to include the two-point fire modeling approach, as indicated in response to RAI 03.c contained in Exelon letter dated May 7, 2018.
4	A review will be conducted of potentially vulnerable fire wrap configurations to identify which are subject to mechanical damage. The fire PRA will be updated to ensure that fire wrap configurations are not credited in fire scenarios that could subject it to mechanical damage, as indicated in response to RAI 03.d contained in Exelon letter dated May 7, 2018.

Peach Bottom 50.69 PRA Implementation Items	
Item #	Description
5	The categorization process includes the fire PRA sensitivity studies from NEI 00-04 Table 5-3 which includes the sensitivity study to remove all credit for manual suppression. Another fire PRA sensitivity study will be performed as part of the categorization process that assumes credit for immediate manual suppression, as indicated in response to RAI 04.a contained in Exelon letter dated May 7, 2018.
6	Sources of uncertainty associated with implementation of FLASHCAT in the fire PRA introduced through use of generic parameters from NUREG-7010 Vol. 1 and weighted averages of parameters for cables located within the physical analysis units in which the scenarios implementing the FLASHCAT model were located will be removed by basing values for these parameters (mass per unit length and plastic mass fraction) on the scenario specific set of cables that are located within the cable trays analyzed using the FLASHCAT model, as indicated in response to RAI 04.a and RAI 04.d contained in Exelon letter dated May 7, 2018.
7	The NUREG-1921 (Reference 14) methodology will be applied in identifying undesired operator actions and will be used to incorporate any identified actions into the fire PRA. A focused-scope peer review of the application of this methodology will be performed, and any new F&Os resulting from the focused scope review will be resolved, as indicated in response to RAI 05.a contained in Exelon letter dated May 7, 2018.
8	The fire PRA model will be updated to address breaker coordination in non-safety related power supplies credited in the model by assuming failure of the power supply, including accounting for opening of the power supply upstream breaker that may occur due to the potential lack of coordination between it and the downstream breaker associated with the damaged power cable, when the power cable within the circuits of concern are identified to be damaged by fire scenarios, or additional analysis will be performed to determine that circuits are coordinated, as indicated in response to RAI 05.b contained in Exelon letter dated May 7, 2018.
9	The PRA model will be updated to account for the requirement for two EDG cooling fans during periods when the outdoor temperatures at Peach Bottom are above the design temperature of 80° F, as indicated in response to RAI 08.a contained in Exelon letter dated May 7, 2018.
10	The pipe rupture frequencies will be updated in the internal flooding PRA to the most recent EPRI pipe rupture frequencies, as indicated in Exelon's letter dated August 30, 2017.
11	Credit for core melt arrest in-vessel at high reactor pressure vessel (RPV) pressure conditions will be removed from the internal events PRA model, as indicated in Exelon's letter dated August 30, 2017.
12	In support of the station's continuing efforts to assess tornado missile protection hazards in response to RIS 2015-06, Exelon is requesting additional time to develop an appropriate response to the issues described in the above question. Exelon will provide a supplemental response to this RAI question with a response date currently expected to be no later than June 6, 2018 (RAI 10.c).

Peach Bottom 50.69 PRA Implementation Items	
Item #	Description
13	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 (RAI 15).

ATTACHMENT 3

License Amendment Request

Peach Bottom Atomic Power Station, Units 2 and 3
Docket Nos. 50-277 and 50-278

Response to Request for Additional Information
Application to Adopt 10 CFR 50.69, "Risk-informed categorization and treatment of structures, systems, and components for nuclear power plants"

Markup of Proposed Renewed Facility Operating License (RFOL) Pages

Unit 2 RFOL Pages
7g

Unit 3 RFOL Pages
7g

2. Level 1 performance criteria.

3. The methodology for establishing the RSD strain limits used for the Level 1 and Level 2 performance.

(e) The results of the power ascension testing to verify the continued structural integrity of the steam dryer shall be submitted to the NRC staff in a report in accordance with 10 CFR 50.4. The report shall include a final load definition and stress report of the steam dryer, including the results of a complete re-analysis using the end-to-end B/Us determined at EPU conditions and a comparison of predicted and measured pressures and strains (RMS levels and spectra) on the RSD. The report shall be submitted within 90 days of the completion of EPU power ascension testing for Peach Bottom Unit 2.

(f) During the first two scheduled refueling outages after reaching EPU conditions, a visual inspection shall be conducted of the steam dryer as described in the inspection guidelines contained in WCAP-17635-P.

(g) The results of the visual inspections of the steam dryer shall be submitted to the NRC staff in a report in accordance with 10 CFR 50.4. The report shall be submitted within 90 days following startup from each of the first two respective refueling outages.

(h) Within 6 months following completion of the second refueling outage, after the implementation of the EPU, the licensee shall submit a long-term steam dryer inspection plan based on industry operating experience along with the baseline inspection results.

The license condition described above shall expire: (1) upon satisfaction of the requirements in paragraphs (f) and (g), provided that a visual inspection of the steam dryer does not reveal any new unacceptable flaw(s) or unacceptable flaw growth that is due to fatigue, and; (2) upon satisfaction of the requirements specified in paragraph (h).

(16) Maximum Extended Load Line Limit Analysis Plus (MELLLA+) Special Consideration

The licensee shall not operate the facility within the MELLLA+ operating domain with a feedwater heater out of service resulting in more than a 10°F reduction in feedwater temperature below the design feedwater temperature.

INSERT 1

(e) The results of the power ascension testing to verify the continued structural integrity of the steam dryer shall be submitted to the NRC staff in a report in accordance with 10 CFR 50.4. The report shall include a final load definition and stress report of the steam dryer, including the results of a complete re-analysis using the end-to-end B/Us from Peach Bottom Unit 2 benchmarking at EPU conditions. The report shall be submitted within 90 days of the completion of EPU power ascension testing for Peach Bottom Unit 3.

(f) During the first two scheduled refueling outages after reaching EPU conditions, a visual inspection shall be conducted of the steam dryer as described in the inspection guidelines contained in WCAP-17635-P.

(g) The results of the visual inspections of the steam dryer shall be submitted to the NRC staff in a report in accordance with 10 CFR 50.4. The report shall be submitted within 90 days following startup from each of the first two respective refueling outages.

(h) Within 6 months following completion of the second refueling outage, after the implementation of the EPU, the licensee shall submit a long-term steam dryer inspection plan based on industry operating experience along with the baseline inspection results.

The license condition described above shall expire: (1) upon satisfaction of the requirements in paragraphs (f) and (g), provided that a visual inspection of the steam dryer does not reveal any new unacceptable flaw(s) or unacceptable flaw growth that is due to fatigue, and; (2) upon satisfaction of the requirements specified in paragraph (h).

(16) Maximum Extended Load Line Limit Analysis Plus (MELLLA+) Special Consideration

The licensee shall not operate the facility within the MELLLA+ operating domain with a feedwater heater out of service resulting in more than a 10°F reduction in feedwater temperature below the design feedwater temperature.

INSERT 1

3. This renewed license is subject to the following conditions for the protection of the environment:

A. To the extent matters related to thermal discharges are treated therein, operation of Peach Bottom Atomic Power Station, Unit No. 3, will be governed by NPDES Permit No. PA 0009733, as now in effect and as hereafter amended. Questions pertaining to conformance thereto shall be referred to and shall be determined by the NPDES Permit issuing or enforcement authority, as appropriate.

B. In the event of any modification of the NPDES Permit related to thermal discharges or the establishment (or amendment) of alternative effluent limitations established pursuant to Section 316 of the Federal Water Pollution Control Act, the Exelon Generation Company shall inform the NRC and analyze any associated changes in or to the Station, its components, its operation or in the discharge of effluents therefrom. If such change would entail any modification to

License Amendment Request
Peach Bottom Atomic Power Station
Units 2 and 3
Adoption of 10 CFR 50.69

License Condition Mark-ups for Renewed Facility Operating Licenses (RFOL)

Unit 2 – INSERT 1

(17) Adoption of 10 CFR 50.69, "Risk-informed categorization and treatment of structures, systems, and components for nuclear power plants"

In support of implementing License Amendment No. [XXX] permitting the adoption of the provisions of 10 CFR 50.69 for Renewed Facility Operating License No. DPR-44 for Peach Bottom Unit 2, the license is amended to add the following license condition:

Prior to the implementation of the provisions of 10 CFR 50.69, the conditions specified below shall be completed:

a) Exelon is approved to implement 10 CFR 50.69 using the processes for categorization of Risk-Informed Safety Class (RISC)-1, RISC-2, RISC-3, and RISC-4 structures, systems, and components (SSCs) specified in the license amendment dated [DATE] subject to the following condition:

Exelon will complete the implementation items listed in Attachment 2 of Exelon letter to NRC dated May 7, 2018, prior to implementation of 10 CFR 50.69. All issues identified in the attachment will be addressed and any associated changes will be made, focused-scope peer reviews will be performed on changes that are PRA upgrades as defined in the PRA standard (ASME/ANS RA-Sa-2009, as endorsed by RG 1.200, Revision 2), and any findings will be resolved and reflected in the PRA of record prior to implementation of the 10 CFR 50.69 categorization process.

Prior NRC approval, under 10 CFR 50.90, is required for a change to the categorization process specified above (e.g., change from a seismic margins approach to a seismic probabilistic risk assessment approach).

Unit 3 – INSERT 1

(17) Adoption of 10 CFR 50.69, "Risk-informed categorization and treatment of structures, systems, and components for nuclear power plants"

In support of implementing License Amendment No. [XXX] permitting the adoption of the provisions of 10 CFR 50.69 for Renewed Facility Operating License No. DPR-56 for Peach Bottom Unit 3, the license is amended to add the following license condition:

Prior to the implementation of the provisions of 10 CFR 50.69 the conditions specified below shall be completed:

a) Exelon is approved to implement 10 CFR 50.69 using the processes for categorization of Risk-Informed Safety Class (RISC)-1, RISC-2, RISC-3, and RISC-4 structures, systems, and components (SSCs) specified in the license amendment dated [DATE] subject to the following condition:

   Exelon will complete the implementation items listed in Attachment 2 of Exelon letter to NRC dated May 7, 2018, prior to implementation of 10 CFR 50.69. All issues identified in the attachment will be addressed and any associated changes will be made, focused-scope peer reviews will be performed on changes that are PRA upgrades as defined in the PRA standard (ASME/ANS RA-Sa-2009, as endorsed by RG 1.200, Revision 2), and any findings will be resolved and reflected in the PRA of record prior to implementation of the 10 CFR 50.69 categorization process.

   Prior NRC approval, under 10 CFR 50.90, is required for a change to the categorization process specified above (e.g., change from a seismic margins approach to a seismic probabilistic risk assessment approach).