Exelon Generation

200 Exelon Way
Kennett Square, PA 19348
www.exeloncorp.com

10 CFR 50.90
10 CFR 50.69

August 30, 2017

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

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: Application to Adopt 10 CFR 50.69, "Risk-informed categorization and treatment of structures, systems, and components for nuclear power reactors"

In accordance with the provisions of 10 CFR 50.69, and 10 CFR 50.90, "Application for amendment of license, construction permit, or early site permit," Exelon Generation Company, LLC (Exelon) is requesting 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." The provisions of 10 CFR 50.69 allow adjustment of the scope of equipment subject to special treatment controls (e.g., quality assurance, testing, inspection, condition monitoring, assessment, and evaluation). For equipment determined to be of low safety significance, alternative treatment requirements can be implemented in accordance with this regulation. For equipment determined to be of high safety significance, requirements will not be changed or will be enhanced. This allows improved focus on equipment that has safety significance resulting in improved plant safety.

The enclosure to this letter provides the basis for the proposed change to the PBAPS, Units 2 and 3, Renewed Facility Operating Licenses. The categorization process being implemented through this change is consistent with NEI 00-04, "10 CFR 50.69 SSC Categorization Guideline," Revision 0, dated July 2005, which was endorsed by the NRC in Regulatory Guide 1.201, "Guidelines for Categorizing Structures, Systems, and Components in Nuclear Power Plants According to their Safety Significance," Revision 1, dated May 2006. Attachment 1 of the enclosure provides a list of categorization prerequisites. Use of the categorization process on a plant system will only occur after these prerequisites are met.

Exelon intends to submit a separate license amendment request to revise PBAPS, Unit 2 and Unit 3, Technical Specifications to adopt TSTF-505, Revision 1, "Provide Risk Informed Extended Completion Times - RITSTF Initiative 4b," within the next six months using the same Probabilistic Risk Assessment (PRA) model described in the enclosure to this letter. Exelon requests that the NRC coordinate their review of the PRA technical adequacy description in Section 3.2 and 3.3 of this enclosure for both applications. This would reduce the number of Exelon and NRC resources necessary to complete the review of the applications. This request should not be considered a linked requested licensing action (RLA), as the details of the PRA models in each LAR are complete which will allow the NRC staff to independently review and approve each LAR on their own merits without regard to the results from the review of the other.

The proposed changes have been reviewed and approved by the PBAPS Plant Operations Review Committee in accordance with the requirements of the Exelon Quality Assurance Program.

Exelon requests approval of the proposed license amendment by August 30, 2018, with the amendment being implemented within 60 days.

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

This letter contains no regulatory commitments.

If you should have any questions regarding this submittal, please contact Richard Gropp at (610) 765-5557.

I declare under penalty of perjury that the foregoing is true and correct. Executed on this 30th day of August 2017.

Respectfully,

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

Enclosure: Evaluation of the Proposed Change

cc: USNRC Region I, Regional Administrator
USNRC Project Manager, PBAPS
USNRC Senior Resident Inspector, PBAPS
Director, Bureau of Radiation Protection - Pennsylvania Department of Environmental Protection
R.R. Janati, Bureau of Radiation Protection
S.T. Gray, State of Maryland

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1 SUMMARY DESCRIPTION

The proposed amendment would modify the licensing basis for Peach Bottom Atomic Power Station (PBAPS), Units 2 and 3, 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 (SSCs) for Nuclear Power Plants." The provisions of 10 CFR 50.69 allow adjustment of the scope of equipment subject to special treatment controls (e.g., quality assurance, testing, inspection, condition monitoring, assessment, and evaluation). For equipment determined to be of low safety significance, alternative treatment requirements can be implemented in accordance with this regulation. For equipment determined to be of high safety significance, requirements will not be changed or will be enhanced. This allows improved focus on equipment that has safety significance resulting in improved plant safety.

2 DETAILED DESCRIPTION

2.1 CURRENT REGULATORY REQUIREMENTS

The Nuclear Regulatory Commission (NRC) has established a set of regulatory requirements for commercial nuclear reactors to ensure that a reactor facility does not impose an undue risk to the health and safety of the public, thereby providing reasonable assurance of adequate protection to public health and safety. The current body of NRC regulations and their implementation are largely based on a "deterministic" approach.

This deterministic approach establishes requirements for engineering margin and quality assurance in design, manufacture, and construction. In addition, it assumes that adverse conditions can exist (e.g., equipment failures and human errors) and establishes a specific set of design basis events (DBEs). The deterministic approach then requires that the facility include safety systems capable of preventing or mitigating the consequences of those DBEs to protect public health and safety. Those SSCs necessary to defend against the DBEs are defined as "safety-related," and these SSCs are the subject of many regulatory requirements, herein referred to as "special treatments," designed to ensure that they are of high quality and high reliability, and have the capability to perform during postulated design basis conditions. Treatment includes, but is not limited to, quality assurance, testing, inspection, condition monitoring, assessment, evaluation, and resolution of deviations. The distinction between "treatment" and "special treatment" is the degree of NRC specification as to what must be implemented for particular SSCs or for particular conditions. Typically, the regulations establish the scope of SSCs that receive special treatment using one of three different terms: "safety-related," "important to safety," or "basic component." The terms "safety-related "and "basic component" are defined in the regulations, while "important to safety," used principally in the general design criteria (GDC) of Appendix A to 10 CFR Part 50, is not explicitly defined.

2.2 REASON FOR PROPOSED CHANGE

A probabilistic approach to regulation enhances and extends the traditional deterministic approach by allowing consideration of a broader set of potential challenges to safety, providing a logical means for prioritizing these challenges based on safety significance, and allowing consideration of a broader set of resources to defend against these challenges. In contrast to the deterministic approach, Probabilistic Risk Assessments (PRAs) address credible initiating events by assessing the event frequency. Mitigating system reliability is then assessed, including the potential for common cause failures. The probabilistic approach to regulation is an extension and enhancement of traditional regulation by considering risk in a comprehensive manner.

To take advantage of the safety enhancements available through the use of PRA, in 2004 the NRC published a new regulation, 10 CFR 50.69. The provisions of 10 CFR 50.69 allow adjustment of the scope of equipment subject to special treatment controls (e.g., quality assurance, testing, inspection, condition monitoring, assessment, and evaluation). For equipment determined to be of low safety significance, alternative treatment requirements can be implemented in accordance with the regulation. For equipment determined to be of high safety significance, requirements will not be changed or will be enhanced. This allows improved focus on equipment that has safety significance resulting in improved plant safety.

The rule contains requirements on how a licensee categorizes SSCs using a risk-informed process, adjusts treatment requirements consistent with the relative significance of the SSC, and manages the process over the lifetime of the plant. A risk-informed categorization process is employed to determine the safety significance of SSCs and place the SSCs into one of four risk-informed safety class (RISC) categories. The determination of safety significance is performed by an integrated decision-making process, as described by NEI 00-04, "10 CFR 50.69 SSC Categorization Guideline" (Reference 1), which uses both risk insights and traditional engineering insights. The safety functions include the design basis functions, as well as functions credited for severe accidents (including external events). Special or alternative treatment for the SSCs is applied as necessary to maintain functionality and reliability, and is a function of the SSC categorization results and associated bases. Finally, periodic assessment activities are conducted to make adjustments to the categorization and/or treatment processes as needed so that SSCs continue to meet all applicable requirements.

The rule does not allow for the elimination of SSC functional requirements or allow equipment that is required by the deterministic design basis to be removed from the facility. Instead, the rule enables licensees to focus their resources on SSCs that make a significant contribution to plant safety. For SSCs that are categorized as high safety significant, existing treatment requirements are maintained or enhanced. Conversely, for SSCs that do not significantly contribute to plant safety on an individual basis, the rule allows an alternative risk-informed approach to treatment that provides reasonable, though reduced, level of confidence that these SSCs will satisfy functional requirements.

Implementation of 10 CFR 50.69 will allow Exelon to improve focus on equipment that has safety significance resulting in improved plant safety.

2.3 DESCRIPTION OF THE PROPOSED CHANGE

Exelon proposes the addition of the following condition to the renewed facility operating licenses of PBAPS, Units 2 and 3, to document the NRC's approval of the use of 10 CFR 50.69.

> 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].
>
> 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).

3 TECHNICAL EVALUATION

10 CFR 50.69 specifies the information to be provided by a licensee requesting adoption of the regulation. This request conforms to the requirements of 10 CFR 50.69(b)(2), which states:

> A licensee voluntarily choosing to implement this section shall submit an application for license amendment under § 50.90 that contains the following information:
>
> (i) A description of the process for categorization of RISC–1, RISC–2, RISC–3 and RISC–4 SSCs.
>
> (ii) 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 (including the plant-specific probabilistic risk assessment (PRA), margins-type approaches, or other systematic evaluation techniques used to evaluate severe accident vulnerabilities) are adequate for the categorization of SSCs.
>
> (iii) Results of the PRA review process conducted to meet § 50.69(c)(1)(i).
>
> (iv) A description of, and basis for acceptability of, the evaluations to be conducted to satisfy § 50.69(c)(1)(iv). The evaluations must include the effects of common cause interaction susceptibility, and the potential impacts from known degradation mechanisms for both active and passive functions, and address internally and externally initiated events and plant operating modes (e.g., full power and shutdown conditions).

Each of these submittal requirements is addressed in the proceeding sections.

Exelon intends to submit a separate license amendment request to revise technical specifications to adopt TSTF-505, Revision 1, "Provide Risk Informed Extended Completion Times - RITSTF Initiative 4b," for PBAPS, Units 2 and 3, within the next six months using the same PRA model described in this enclosure. Exelon requests that the NRC coordinate their review of the PRA technical adequacy description in Section 3.2 and 3.3 of this enclosure for both applications. This would reduce the number of Exelon and NRC resources necessary to complete the review of the applications. This request should not be considered a linked requested licensing action (RLA), as the details of the PRA models in each LAR are complete which will allow the NRC staff to independently review and approve each LAR on their own merits without regard to the results from the review of the other.

3.1 CATEGORIZATION PROCESS DESCRIPTION (10 CFR 50.69(b)(2)(i))

3.1.1 Overall Categorization Process

Exelon will implement the risk categorization process in accordance with NEI 00-04, Revision 0, as endorsed by RG 1.201, "Guidelines for Categorizing Structures, Systems, and Components in Nuclear Power Plants According to their Safety Significance," (Reference 2). NEI 00-04, Section 1.5, states "Due to the varying levels of uncertainty and degrees of conservatism in the spectrum of risk contributors, the risk significance of SSCs is assessed separately from each of five risk perspectives and used to identify SSCs that are potentially safety- significant." Separate evaluation is appropriate to avoid reliance on a combined result that may mask the results of individual risk contributors.

The following are clarifications to be applied to the NEI 00-04 categorization process:

- The Integrated Decision-making Panel (IDP) will be composed of a group of at least five experts who collectively have expertise in plant operation, design engineering, system engineering, safety analysis, and probabilistic risk assessment. At least three members of the IDP will have a minimum of five years of experience at the plant, and there will be at least one member of the IDP who has a minimum of three years of experience in modeling and updating of the plant-specific PRA.
- The IDP will be trained in the specific technical aspects and requirements related to the categorization process. Training will address at a minimum the purpose of the categorization; present treatment requirements for SSCs including requirements for: design basis events; PRA fundamentals; details of the plant specific PRA including the modeling, scope, and assumptions, the interpretation of risk importance measures, and the role of sensitivity studies and the change-in-risk evaluations; and the defense-in-depth philosophy and requirements to maintain this philosophy.
- The decision criteria for the IDP for categorizing SSCs as safety significant or low safety-significant pursuant to § 50.69(f)(1) will be documented in Exelon procedures. Decisions of the IDP will be arrived at by consensus. Differing opinions will be documented and resolved, if possible. If a resolution cannot be achieved concerning the safety significance of an SSC, then the SSC will be classified as safety-significant.
- Passive characterization will be performed using the processes described in Section 3.1.2 of this enclosure.

- An unreliability factor of 3 will be used for the sensitivity studies described in Section 8 of NEI 00-04. The factor of 3 was chosen as it is representative of the typical error factor of basic events used in the PRA model.
- Exelon will require that if any SSC is identified as high safety significant (HSS) from either the integrated PRA component safety significance assessment (Section 5 of NEI 00-04) or the defense-in-depth assessment (Section 6 of NEI 00-04), the associated system function(s) would be identified as HSS.
- Once a system function is identified as HSS, then all the components that support that function are preliminary HSS. The Integrated Decision-making Panel (IDP) must intervene to assign any of these HSS Function components to LSS.
- With regard to the criteria that considers whether the active function is called out or relied upon in the plant Emergency/Abnormal Operating Procedures, PBAPS will not take credit for alternate means unless the alternate means are proceduralized and included in Licensed Operator training.

The risk analysis being implemented for each hazard is described:

- Internal Event Risks: Internal events including internal flooding PRA model PB214A5 (Unit 2) and PB314A5 (Unit 3), May 2017.
- Fire Risks: Fire PRA model versions PB214A5F0 (Unit 2) and PB314A5F0 (Unit 3), May 2017.
- Seismic Risks: Success Path Component List (SPCL) from the IPEEE seismic analysis accepted by NRC SER dated November 22, 1999, TAC NOS. M83657 and M83658 (Reference 3).
- Other External Risks (e.g., tornados, external floods, etc.): Using the IPEEE screening process as approved by NRC SER dated November 22, 1999, TAC NOS. M83657 and M83658 (Reference 3). The other external hazards were determined to be insignificant contributors to plant risk.
- Low Power and Shutdown Risks: Qualitative defense-in-depth (DID) shutdown model for shutdown configuration risk management (CRM) based on the framework for DID provided in NUMARC 91-06, "Guidance for Industry Actions to Assess Shutdown Management" (Reference 4), which provides guidance for assessing and enhancing safety during shutdown operations.

A change to the categorization process that is outside the bounds specified above (e.g., change from a seismic margins approach to a seismic probabilistic risk assessment approach) will not be used without prior NRC approval. The SSC categorization process documentation will include the following elements:

1. Program procedures used in the categorization
2. System functions, identified and categorized with the associated bases
3. Mapping of components to support function(s)
4. PRA model results, including sensitivity studies
5. Hazards analyses, as applicable
6. Passive categorization results and bases
7. Categorization results including all associated bases and RISC classifications
8. Component critical attributes for HSS SSCs
9. Results of periodic reviews and SSC performance evaluations
10. IDP meeting minutes and qualification/training records for the IDP members

3.1.2 Passive Categorization Process

For the purposes of 10 CFR 50.69 categorization, passive components are those components that have a pressure retaining function. Passive components and the passive function of active components will be evaluated using the Risk-Informed Repair/Replacement Activities (RI-RRA) methodology consistent with the Safety Evaluation (SE) by the Office of Nuclear Reactor Regulation "Request tor 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 In-service Inspection Intervals', dated April 22, 2009 (ML090930246) (Reference 5).

The RI-RRA methodology is a risk-informed safety classification and treatment program for repair/replacement activities (RI-RRA methodology) for pressure retaining items and their associated supports. In this method, the component failure is assumed with a probability of 1.0 and only the consequence evaluation is performed. It additionally applies deterministic considerations (e.g., defense in depth, safety margins) in determining safety significance. Component supports are assigned the same safety significance as the highest passively ranked component within the bounds of the associated analytical pipe stress model.

The requirements of 10 CFR 50.69 are consistent with ANO-2 RI-RRA License Amendment as the rule does not remove the repair and replacement provisions of the ASME Code required by § 50.55a(g) for ASME Class 1 SSCs, even if they are categorized as RISC-3, since those SSCs constitute principal fission product barriers as part of the reactor coolant system or containment. This is further clarified in the rule's Statement of Considerations. However, since the scope of 10 CFR 50.69 addresses additional requirements, this methodology will be applied to determine the safety significance of ASME Class 1 SSCs, some of which may be evaluated to be RISC-3. The

ASME classification of the SSC does not impact the methodology as it only evaluates the consequence of a rupture of the SSC's pressure boundary. As stated in the ANO SE, "categorizing solely based on consequence which measures the safety significance of the pipe given that it ruptures is conservative compared to including the rupture frequency in the categorization and the categorization will not be affected by changes in frequency arising from changes to the treatment." Therefore, this methodology is appropriate to apply to ASME Class 1 SSCs, as the consequence evaluation and deterministic considerations are independent of the ASME classification when determining the SSCs safety significance and will maintain this acceptable level of conservatism.

The use of this method was previously approved to be used for a 10 CFR 50.69 application by NRC in the final Safety Evaluation for Vogtle Electric Generating Plant (Vogtle) dated December 17, 2014 (Reference 6). The RI-RRA method as approved for use at Vogtle for 10 CFR 50.69 does not have any plant specific aspects and is generic. It relies on the conditional core damage and large early release probabilities associated with postulated ruptures. Safety significance is generally measured by the frequency and the consequence of the event. However, this RI-RRA process categorizes components solely based on consequence, which measures the safety significance of the passive component given that it ruptures. This approach is conservative compared to including the rupture frequency in the categorization as this approach will not allow the categorization of SSCs to be affected by any changes in frequency due to changes in treatment. Therefore, the RI-RRA methodology for passive categorization is acceptable and appropriate for use at PBAPS for 10 CFR 50.69.

3.2 TECHNICAL ADEQUACY EVALUATION (10 CFR 50.69(b)(2)(ii))

The following sections demonstrate that the quality and level of detail of the processes used in categorization of SSCs are adequate. All the PRA models described below have been peer reviewed and there are no PRA upgrades that have not been peer reviewed.

3.2.1 Internal Events and Internal Flooding

The PBAPS categorization process for the internal events and flooding hazard will use the peer reviewed plant-specific PRA model. The Exelon risk management process ensures that the PRA model used in this application reflects the as-built and as-operated plant for each of the PBAPS units. Attachment 2 of this enclosure identifies the applicable internal events and internal flooding PRA models.

3.2.2 Fire Hazards

The PBAPS categorization process for fire hazards will use a peer reviewed plant-specific fire PRA model. The internal Fire PRA model was developed consistent with

NUREG/CR-6850 and only utilizes methods previously accepted by the NRC. The Exelon risk management process ensures that the PRA model used in this application reflects the as-built and as-operated plant for each of the PBAPS units. Attachment 2 of this enclosure identifies the applicable Fire PRA model.

3.2.3 Seismic Hazards

The PBAPS categorization process will use the seismic margins analysis (SMA) performed for the Individual Plant Evaluation-External Events (IPEEE) in response to GL 88-20 (Reference 7) for evaluation of safety significance related to seismic hazards. No plant specific approaches were utilized in development of the SMA. The NEI 00-04 approved use of the SMA safe shutdown equipment list (SSEL) (also called the success path component list, or SPCL at PBAPS) as a screening process identifies all system functions and associated SSCs that are involved in the seismic margin success path as HSS. Since the analysis is being used as a screening tool, importance measures are not used to determine safety significance. The NEI 00-04 approach using the SPCL would identify credited equipment as HSS regardless of their capacity, frequency of challenge or level of functional diversity.

An evaluation was performed of the as-built, as-operated plant against the SMA SPCL. The evaluation was a comparison of the as-built, as-operated plant to the plant configuration originally assessed by the SMA. Differences were reviewed to identify any potential impacts to the equipment credited on the SPCL. Appropriate changes to the credited equipment were identified and documented. This documentation is available for audit. The Exelon risk management program ensures that future changes to the plant will be evaluated to determine their impact on the SMA and risk categorization process.

3.2.4 Other External Hazards

The PBAPS categorization process will use screening results from the Individual Plant Evaluation of External Events (IPEEE) in response to GL 88-20 (Reference 7) for evaluation of safety significance related to the following other external hazards:

- High Winds
- External Flooding
- Transportation and Nearby Facility Accidents
- Other External Initiating Events (i.e., other "HFO" Events)

All SSCs credited in other IPEEE external hazards are considered HSS. All other external hazards were screened from applicability to PBAPS, Units 2 and 3, per a plant-specific evaluation in accordance with GL 88-20 (Reference 3) and updated to use the criteria in ASME PRA Standard RA-Sa-2009 (Reference 8). Attachment 4 of this enclosure provides a summary of the other external hazards screening results.

Attachment 5 of this enclosure provides a summary of the progressive screening approach for external hazards.

3.2.5 Low Power & Shutdown

The PBAPS categorization process will use the shutdown safety management plan described in NUMARC 91-06, for evaluation of safety significance related to low power and shutdown conditions.

3.2.6 PRA Maintenance and Updates

The Exelon risk management process ensures that the applicable PRA models used in this application continue to reflect the as-built and as-operated plant for each of the PBAPS units. The process delineates the responsibilities and guidelines for updating the PRA models, and includes criteria for both regularly scheduled and interim PRA model updates. The process includes provisions for monitoring potential areas affecting the PRA models (e.g., due to changes in the plant, errors or limitations identified in the model, industry operational experience) for assessing the risk impact of unincorporated changes, and for controlling the model and associated computer files. The process will assess the impact of these changes on the plant PRA model in a timely manner but no longer than once every two refueling outages. If there is a significant impact on the PRA model, the SSC categorization will be re-evaluated.

In addition, Exelon will implement a process that addresses the requirements in NEI 00-04, Section 11, "Program Documentation and Change Control." The process will review the results of periodic and interim updates of the plant PRA that may affect the results of the categorization process. If the results are affected, adjustments will be made as necessary to the categorization or treatment processes to maintain the validity of the processes. In addition, any PRA model upgrades will be peer reviewed prior to implementing those changes in the PRA model used for categorization.

3.2.7 PRA Uncertainty Evaluations

Uncertainty evaluations associated with any applicable baseline PRA model(s) used in this application were evaluated during the assessment of PRA technical adequacy and confirmed through the self-assessment and peer review processes as discussed in Section 3.3 of this enclosure.

Uncertainty evaluations associated with the risk categorization process are addressed using the processes discussed in Section 8 of NEI 00-04 and in the prescribed sensitivity studies discussed in Section 5 of NEI 00-04.

In the overall risk sensitivity studies, Exelon will utilize a factor of 3 to increase the unavailability or unreliability of LSS components consistent with that approved for

Vogtle in Reference 6. Consistent with the NEI 00-04 guidance, Exelon will perform both an initial sensitivity study and a cumulative sensitivity study. The initial sensitivity study applies to the system that is being categorized. In the cumulative sensitivity study, the failure probabilities (unreliability and unavailability, as appropriate) of all LSS components modeled in PRAs for all systems that have been categorized are increased by a factor of 3. This sensitivity study together with the periodic review process assures that the potential cumulative risk increase from the categorization is maintained acceptably low. The performance monitoring process monitors the component performance to ensure that potential increases in failure rates of categorized components are detected and addressed before reaching the rate assumed in the sensitivity study.

Sources of model uncertainty and related assumptions have been identified for the PBAPS PRA models using the guidance of NUREG-1855 (Reference 9) and EPRI TR-1016737, "Treatment of Parameter and Model Uncertainty for Probabilistic Risk Assessments" (Reference 10).

The detailed process of identifying, characterizing and qualitative screening of model uncertainties is found in Section 5.3 of NUREG-1855 and Section 3.1.1 of EPRI TR-1016737. The process in these references was mostly developed to evaluate the uncertainties associated with the internal events PRA model; however, the approach can be applied to other types of hazard groups.

The list of assumptions and sources of uncertainty have been reviewed to identify those which would be significant for the evaluation of this application. If the PBAPS PRA model uses non-conservative treatments, or uses methods not commonly accepted, the underlying assumption or source of uncertainty was reviewed to determine its impact on this application. Only those assumptions or sources of uncertainty that could significantly impact the categorization risk calculations were considered key for this application.

Key PBAPS PRA model specific assumptions and sources of uncertainty for this application have been identified and dispositioned in Attachment 6. The conclusion of this review is that no additional sensitivity analyses are required to address PBAPS PRA model specific assumptions or sources of uncertainty.

3.3 PRA REVIEW PROCESS RESULTS (10 CFR 50.69(b)(2)(iii))

The PRA models described in Section 3.2 above have been assessed against RG 1.200, "An Approach for Determining the Technical Adequacy of Probabilistic Risk Assessment Results for Risk-Informed Activities," Revision 2 (Reference 11), consistent with NRC RIS 2007-06. The internal events PRA model was subject to a self-assessment and a full-scope peer review conducted in November 2010.

The fire PRA model was subject to a self-assessment and a full-scope peer review conducted in November 2011.

Closed findings were reviewed and closed using the process documented in Appendix X to NEI 05-04, NEI 07-12 and NEI 12-13, "Close-out of Facts and Observations" (F&Os) (Reference 12) as accepted by NRC in the staff memorandum dated May 3, 2017 (ML17079A427) (Reference 13). The results of this review have been documented and are available for NRC audit. The NRC observed the closure review of open PRA Facts and Observations (F&Os) for PBAPS, Units 2 and 3, in November 2016.

Attachment 3 of this enclosure provides a summary disposition of:

- Open items and disposition from the PBAPS RG 1.200 self-assessment.
- Open findings and disposition of the PBAPS PRA peer reviews.

This information demonstrates that the PRA is of sufficient quality and level of detail to support the categorization process, and has been subjected to a peer review process assessed against a standard or set of acceptance criteria that is endorsed by the NRC as required 10 CFR 50.69(c)(1)(i).

3.4 RISK EVALUATIONS (10 CFR 50.69(b)(2)(iv))

The PBAPS 10 CFR 50.69 categorization process will implement the guidance in NEI 00-04. The overall risk evaluation process described in the NEI guidance addresses both known degradation mechanisms and common cause interactions, and meets the requirements of §50.69(b)(2)(iv). Sensitivity studies described in NEI 00-04, Section 8, will be used to confirm that the categorization process results in acceptably small increases to core damage frequency (CDF) and large early release frequency (LERF). The failure rates for equipment and initiating event frequencies used in the PRA include the quantifiable impacts from known degradation mechanisms, as well as other mechanisms (e.g., design errors, manufacturing deficiencies, human errors, etc.). Subsequent performance monitoring and PRA updates required by the rule will continue to capture this data, and provide timely insights into the need to account for any important new degradation mechanisms.

4 REGULATORY EVALUATION

4.1 APPLICABLE REGULATORY REQUIREMENTS/CRITERIA

The following NRC requirements and guidance documents are applicable to the proposed change:

- The regulations at 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."
- NRC Regulatory Guide 1.201, "Guidelines for Categorizing Structures, Systems, and Components in Nuclear Power Plants According to their Safety Significance," Revision 1, May 2006.
- Regulatory Guide 1.174, "An Approach for Using Probabilistic Risk Assessment in Risk-Informed Decisions on Plant-Specific Changes to the Licensing Basis," Revision 2, April 2015.
- Regulatory Guide 1.200, "An Approach for Determining the Technical Adequacy of Probabilistic Risk Assessment Results for Risk-Informed Activities," Revision 2, U.S. Nuclear Regulatory Commission, March 2009.

The proposed change is consistent with the applicable regulations and regulatory guidance.

4.2 NO SIGNIFICANT HAZARDS CONSIDERATION ANALYSIS

Exelon proposes to modify the licensing basis for Peach Bottom Atomic Power Station (PBAPS), Units 2 and 3, by the addition of a license condition to allow for the voluntary implementation of the provisions of Title 10 of the Code of Federal Regulations (10 CFR), Part 50.69, "Risk-informed categorization and treatment of structures, systems, and components for nuclear power reactors." The provisions of 10 CFR 50.69 allow adjustment of the scope of equipment subject to special treatment controls (e.g., quality assurance, testing, inspection, condition monitoring, assessment, and evaluation). For equipment determined to be of low safety significance, alternative treatment requirements can be implemented in accordance with this regulation. For equipment determined to be of high safety significance, requirements will not be changed or will be enhanced. This allows improved focus on equipment that has safety significance resulting in improved plant safety.

Exelon has evaluated whether or not a significant hazards consideration is involved with the proposed amendments by focusing on the three standards set forth in 10 CFR 50.92, "Issuance of amendment," as discussed below:

1. Does the proposed change involve a significant increase in the probability or consequences of an accident previously evaluated?

   Response: No.

   The proposed change will permit the use of a risk-informed categorization process to modify the scope of SSCs subject to NRC special treatment requirements and to implement alternative treatments per the regulations. The process used to evaluate SSCs for changes to NRC special treatment requirements and the use of alternative requirements ensures the ability of the SSCs to perform their design function. The potential change to special treatment requirements does not change the design and operation of the SSCs. As a result, the proposed change does not significantly affect any initiators to accidents previously evaluated or the ability to mitigate any accidents previously evaluated. The consequences of the accidents previously evaluated are not affected because the mitigation functions performed by the SSCs assumed in the safety analysis are not being modified. The SSCs required to safely shut down the reactor and maintain it in a safe shutdown condition following an accident will continue to perform their design functions.

   Therefore, the proposed change does not involve a significant increase in the probability or consequences of an accident previously evaluated.

2. Does the proposed change create the possibility of a new or different kind of accident from any accident previously evaluated?

   Response: No.

   The proposed change will permit the use of a risk-informed categorization process to modify the scope of SSCs subject to NRC special treatment requirements and to implement alternative treatments per the regulations. The proposed change does not change the functional requirements, configuration, or method of operation of any SSC. Under the proposed change, no additional plant equipment will be installed.

   Therefore, the proposed change does not create the possibility of a new or different kind of accident from any accident previously evaluated.

3. Does the proposed change involve a significant reduction in a margin of safety?

   Response: No.

   The proposed change will permit the use of a risk-informed categorization process to modify the scope of SSCs subject to NRC special treatment requirements and to implement alternative treatments per the regulations. The

proposed change does not affect any Safety Limits or operating parameters used to establish the safety margin. The safety margins included in analyses of accidents are not affected by the proposed change. The regulation requires that there be no significant effect on plant risk due to any change to the special treatment requirements for SSCs and that the SSCs continue to be capable of performing their design basis functions, as well as to perform any beyond design basis functions consistent with the categorization process and results.

Therefore, the proposed change does not involve a significant reduction in a margin of safety.

Based on the above, Exelon concludes that the proposed change presents no significant hazards consideration under the standards set forth in 10 CFR 50.92(c), and, accordingly, a finding of "no significant hazards consideration" is justified.

4.3 CONCLUSIONS

In conclusion, based on the considerations discussed above, (1) there is reasonable assurance that the health and safety of the public will not be endangered by operation in the proposed manner, (2) such activities will be conducted in compliance with the Commission's regulations, and (3) the issuance of the amendments will not be inimical to the common defense and security or to the health and safety of the public.

5 ENVIRONMENTAL CONSIDERATION

A review has determined that the proposed amendments would change a requirement with respect to installation or use of a facility component located within the restricted area, as defined in 10 CFR 20, or would change an inspection or surveillance requirement. However, the proposed amendments do not involve (i) a significant hazards consideration, (ii) a significant change in the types or a significant increase in the amounts of any effluents that may be released offsite, or (iii) a significant increase in individual or cumulative occupational radiation exposure. Accordingly, the proposed amendments meet the eligibility criterion for categorical exclusion set forth in 10 CFR 51.22(c)(9). Therefore, pursuant to 10 CFR 51.22(b), no environmental impact statement or environmental assessment need be prepared in connection with the proposed amendments.

6 REFERENCES

1. NEI 00-04, "10 CFR 50.69 SSC Categorization Guideline," Revision 0, Nuclear Energy Institute, July 2005.
2. NRC Regulatory Guide 1.201, "Guidelines for Categorizing Structures, Systems, and Components in Nuclear Power Plants According to their Safety Significance," Revision 1, May 2006.
3. Review of Peach Bottom Atomic Power Station, Units 2 and 3, Individual Plant Examination of External Events (IPEEE) Submittal, (TAC NOS. M83657 and M83658), US Nuclear Regulatory Commission, November 22, 1999. [This includes a safety evaluation report (SER)].
4. NUMARC 91-06, "Guidelines for Industry Actions to Assess Shutdown Management," December, 1991.
5. ANO-2 SE, Safety Evaluation by the Office of Nuclear Reactor Regulation Request for Alternative AN02-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 In-service Inspection Intervals, dated April 22, 2009.
6. Vogtle Electric Generating Plant, Units 1 and 2 -Issuance of Amendments Re: Use of 10 CFR 50.69 (TAC NOS. ME9472 AND ME9473), December 17, 2014.
7. Generic Letter 88-20, "Individual Plant Examination of External Events (IPEEE) for Severe Accident Vulnerabilities - 10 CFR 50.54(f), Supplement 4," USNRC, June 1991.
8. Addendum A, 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
9. NUREG-1855, Guidance on the Treatment of Uncertainties Associated with PRAs in Risk-Informed Decision Making, Revision 1, March 2017.
10. EPRI TR 1016737, "Treatment of Parameter and Model Uncertainty for Probabilistic Risk Assessments," 2008.
11. Regulatory Guide 1.200, "An Approach for Determining the Technical Adequacy of Probabilistic Risk Assessment Results for Risk-Informed Activities," Revision 2, US Nuclear Regulatory Commission, March 2009.
12. NEI Letter to USNRC, "Final Revision of Appendix X to NEI 05-04/07-12/12-16, Close-Out of Facts and Observations (F&Os)," February 21, 2017, Accession Number ML17086A431.

13. USNRC Letter to Mr. Greg Krueger (NEI), "U.S. Nuclear Regulatory Commission Acceptance on Nuclear Energy Institute Appendix X to Guidance 05-04, 7-12, and 12-13, Close Out of Facts and Observations (F&Os)," May 3, 2017, Accession Number ML17079A427.
14. Peach Bottom Atomic Power Station – Units 2 & 3 Flood Hazard Reevaluation Report (FHRR), RS-15-163, NRC Docket Nos. 50-277 and 50-278, August 12, 2015.
15. Peach Bottom Atomic Power Station UFSAR, Revision 26, April, 2017.
16. Philadelphia Electric Company, Peach Bottom Atomic Power Station Individual Plant Examination for External Events (IPEEE), 29 May 1996
17. Licensee Event Report (LER) 2-17-001, March 8, 2017
18. NRC Regulatory Guide 1.91, "Evaluations of Explosions postulated to Occur at Nearby Facilities and on Transportation Routes Near Nuclear Power Plants," Revision 2
19. Peach Bottom Surveillance ST-B-40D-800-2; "Offsite Hazardous Chemical (Toxic Gas) Survey for Control Room Habitability"
20. Peach Bottom calculation PM-1085 Revision 1; "PBAPS CHR Analysis for the Offsite Chemicals Stored at Calpine Site"; June 10, 2016
21. EPRI TR-1013141, "Pipe Rupture Frequencies for Internal Flooding PRAs," Rev. 1, Electric Power Research Institute, Palo Alto, CA, 2006.
22. "Nuclear Power Plant Fire Ignition Frequency and Non-Suppression Probability Estimation Using the Updated Fire Events Database, United States Fire Event Experience Through 2009," NUREG-2169/EPRI 3002002936, U.S. NRC and Electric Power Research Institute, January 2015.
23. "Refining and Characterizing Heat Release Rates From Electrical Enclosures during Fire (RACHELLE-FIRE), Volume 1: Peak Heat Release Rates and Effect of Obstructed Plume," NUREG-2178 Vol. 1/ EPRI 3002005578, U.S. NRC and Electric Power Research Institute, Draft Report for Comment, April 2015.
24. "Joint Assessment of Cable Damage and Quantification of Effects from Fire (JACQUE-FIRE), Volume 2: Expert Elicitation Exercise for Nuclear Power Plant Fire-Induced Electrical Circuit Failure," Final Report, NUREG/CR-7150, Vol. 1, EPRI 3002001989, U.S. NRC and Electric Power Research Institute, May 2014.

Attachment 1: List of Categorization Prerequisites

Exelon will establish procedure(s) prior to the use of the categorization process on a plant system. The procedure(s) will contain the elements/steps listed below.

- Integrated Decision-making Panel (IDP) member qualification requirements
- Qualitative assessment of system functions. System functions are qualitatively categorized as preliminary HSS or LSS based on the seven questions in Section 9 of NEI 00-04 (see Section 3.2 of this enclosure). Any component supporting an HSS function is categorized as preliminary HSS. Components supporting an LSS function are categorized as preliminary LSS.
- Component safety significance assessment. Safety significance of active components is assessed through a combination of PRA and non-PRA methods, covering all hazards. Safety significance of passive components is assessed using a methodology for passive components.
- Assessment of defense-in-depth (DID) and safety margin. Components that are categorized as preliminary LSS are evaluated for their role in providing defense-in-depth and safety margin and, if appropriate, upgraded to HSS.
- Review by the Integrated Decision-making Panel. The categorization results are presented to the IDP for review and approval. The IDP reviews the categorization results and makes the final determination on the safety significance of system functions and components.
- Risk sensitivity study. For PRA-modeled components, an overall risk sensitivity study is used to confirm that the population of preliminary LSS components results in acceptably small increases to core damage frequency (CDF) and large early release frequency (LERF) and meets the acceptance guidelines of RG 1.174.
- Periodic reviews are performed to ensure continued categorization validity and acceptable performance for those SSCs that have been categorized.
- Documentation requirements as discussed in Section 3.1.1 of this enclosure.

Attachment 2: Description of PRA Models used in Categorization

Units	Model	Baseline CDF	Baseline LERF	Comments
2 & 3	**Full Power Internal Events & Internal Flooding** PB214A5 (Unit 2) PB314A5 (Unit 3)	3.4E-06 (Unit 2) 3.0E-06 (Unit 3)	4.9E-07 (Unit 2) 3.8E-07 (Unit 3)	Application-Specific Model (ASM) to the 2014 Full Power Internal Events (FPIE) PRA Model of Record (MOR).
2 & 3	**Fire** PB214A5F0 (Unit 2) PB314A5F0 (Unit 3)	3.8E-05 (Unit 2) 4.1E-05 (Unit 3)	3.0E-06 (Unit 2) 3.3E-06 (Unit 3)	ASM to the Fire Update based on the 2014 FPIE PRA Model.

Attachment 3: Disposition and Resolution of Open Peer Review Findings and Self-Assessment Open Items

Finding Number	Supporting Require-ment(s)	Capability Category (CC)	Finding Description	Disposition for 50.69
2011-1-7	IFSN-A6	Not Met	Note: This SR is modified by the notes in the RG 1.200. Following those notes, this SR can only be judged to be met at CC I. No assessment was done relating to factors such as pipe whip, humidity, condensation, etc., as required by the RG 1.200 notes.	PARTIALLY RESOLVED – OPEN DOCUMENTATION Pipe whip effects were investigated and shown to not be a concern for piping containing moderate energy water sources. Jet impingement effects were also shown to not be a concern for piping encapsulated by aluminum lagging. Humidity effects due to high temperature pipe ruptures were deemed negligible due to the large room volume often surrounding a pipe break, such as in the Turbine Building. High energy piping, such as steam and feedwater, are mainly located in open spaces within the Turbine Building. The PRA model already accounts for steam line and feedwater line break scenarios, which render the Power Conversion System (PCS) equipment unavailable, e.g., condensate and main feedwater pumps, main condenser, etc. Therefore, any additional internal flood event scenario involving high energy lines is already subsumed by these existing initiators in the PRA model. Flood and spray scenarios were also assumed to damage all equipment within the flood area regardless of the damage mechanism. This presents a bounding impact on the risk analysis

Finding Number	Supporting Require-ment(s)	Capability Category (CC)	Finding Description	Disposition for 50.69
				and is independent of any one cause for equipment failure, such as due to humidity, condensation, etc. This is a documentation issue with no impact on this application.
2011-3-1	HR-A1	Not Met	Alignment pre-initiators are included for some risk significant systems (i.e., HPCI, RCIC, LPCS, and SLCS), but these were not included as a result of a review of procedures and practices. Refer to Sections 2.3.3, 4.3, 5.1, and Appendix B of the HRA Notebook (PB-PRA-004).	OPEN The SR that this finding is derived from is related to test and maintenance pre-initiators. The PRA model includes several of these pre-initiators for a number of risk significant systems. These pre-initiators were not derived from a formal review of procedures and practices but were evaluated using the ASEP methodology as described in Appendix B of the HRA Notebook (PB-HRA-004). The basic events modeling these pre-initiators have low failure probabilities. Resolution of this issue will have minimal impact on this application.
2011-3-4	HR-B1	Not Met	The process described in the HRA Notebook (PB-PRA-004) does not establish any rules for screening individual activities. Some System Notebooks (PB-PRA-005) (e.g., HPCI, RCIC, LPCS, SLCS) include pre-initiators and identify appropriate screening rules in Section 6.1.5 but do not identify activities which might	OPEN The SR that this finding is derived from is related to test and maintenance pre-initiators. The PRA model includes several of these pre-initiators for a number of risk significant systems. These pre-initiators were not derived from a formal review of procedures and practices but were evaluated using the ASEP methodology as described in Appendix B of the HRA

Finding Number	Supporting Require-ment(s)	Capability Category (CC)	Finding Description	Disposition for 50.69
			have been screened.	Notebook (PB-HRA-004). The basic events modeling these pre-initiators have low failure probabilities. Resolution of this issue will have minimal impact on this application.
2011-3-6	HR-C2	Met CC II/III	No evidence was found that plant testing procedures were used to define pre-initiator activities that would cause system unavailability or plant trips.	OPEN The SR that this finding is derived from is related to test and maintenance pre-initiators. The PRA model includes several of these pre-initiators for a number of risk significant systems. These pre-initiators were not derived from a formal review of procedures and practices but were evaluated using the ASEP methodology as described in Appendix B of the HRA Notebook (PB-HRA-004). The basic events modeling these pre-initiators have low failure probabilities. Resolution of this issue will have minimal impact on this application.
2011-5-8	HR-D2	Not Met	Table 5.1-4 of the HRA Notebook (PB-PRA-004) includes a number of pre-initiators types (e.g., flow, delta-temperature, steam leak) that are not documented in Table 5.1-2 or Appendix B.	OPEN The SR that this finding is derived from is related to test and maintenance pre-initiators. The PRA model includes several of these pre-initiators for a number of risk significant systems. These pre-initiators were not derived from a formal review of procedures and practices but were evaluated using the ASEP methodology as described in Appendix B of the HRA

Finding Number	Supporting Require-ment(s)	Capability Category (CC)	Finding Description	Disposition for 50.69
				Notebook (PB-HRA-004). The basic events modeling these pre-initiators have low failure probabilities. Resolution of this issue will have minimal impact on this application.
2011-6-3	IE-D2	Met CC I/II/III	There is adequate documentation to meet the SR. The treatment of four categories of LOOP is an improvement. However, there is room for improvement: 1. See F&O written in response to IE-B3 to improve documentation. 2. There are a lot of pages written up to calculate the frequency of Large LOCA, but it does not look like the value is used in the PRA. The documentation can be simplified by just referring to the value used and eliminating the text relating to the unused value. 3. The steam LOCA and liquid LOCA seem to be getting lumped together. It is not clear if these LOCAs are treated in the same manner (i.e., using the same success criteria).	PARTIALLY RESOLVED The following discussion addresses the five items: 1 – This item refers to F&O IE-B3 which involved providing better documentation of initiating event groupings in the Initiating Event Notebook. F&O IE-B3 was considered to be resolved by the 2016 F&O Technical Review Team. 2 – This item refers to Section 3.3.2 of the IE Notebook. Latest LLOCA value of 5.2E-5 is used in the model. The historical text is retained in the notebook and was not simplified (this is acceptable). 3 - IE Notebook – 3.3.2 for the LOCA analysis. Data Notebook Appendix G.4.4 provides the analysis for the fraction of above or below TAF breaks which are modeled under an AND gate next to the respective LOCA initiator. 4 – This item involved updating of the ISLOCA analyses. F&O 2011-2-5 was considered to be

Finding Number	Supporting Require-ment(s)	Capability Category (CC)	Finding Description	Disposition for 50.69
			4. The ISLOCA analysis has not been updated from the IPE days. 5. It might be useful to document why certain events such as the following are excluded from the PRA: Multiple IORV, Multiple SORV, Stuck-open safety valve.	resolved by the 2016 F&O Technical Review Team based on inspection of the new ISLOCA Notebook, PB-PRA-011. Section 2 clearly presents the process and results for identifying and screening ISLOCA pathways. In Section 3, the configuration of the remaining pathways is presented, accounting for valve types, indications, testability, etc. Surveillance test requirements are documented in Table 3-1. Operator actions in response to ISLOCA, including available cues, are addressed in Section 3.3. The frequency calculations (Section 4) are based on fault tree models that consider all these issues. 5 – Spurious safety valve opening is included with medium LOCA, multiple valves are treated with spurious ADS and is included in large LOCA. IORV is modeled as initiating event. Transient induced stuck open valve is also included in event tree modeling. The frequency of two or more relief valves opening is bounded/dominated by spurious initiation of ADS. Among all of the safety relief valves (ADS and non-ADS valves), spurious ADS is the dominant contributor to the LLOCA frequency. To address the F&O, the Initiating Event Notebook (PB-PRA-001) will be updated to add to the spurious ADS discussion.

Finding Number	Supporting Require-ment(s)	Capability Category (CC)	Finding Description	Disposition for 50.69
				As resolution of this finding is a documentation enhancement there will be minimal impact on this application.
2012-1-15	FSS-A5	Not Met	For Cable Room Fires, the assumptions used to analyze the Fire Scenarios are summarized as follows: 1) The events starting $ASD represent the probability that a cabinet fire leaves the cabinet and damages the first tray overhead. 2) Once Damage occurs, since actuation of the CO2 System would result in a control room abandonment, the ASD event tree logic is used for all fires damaging overhead cables - regardless of the cables damage and the extent of damage. No credit for manual suppression without actuation of CO2 is taken. 3) Non-Control Room abandonment fires are analyzed, but are limited to the systems impacted by cables within the cabinet. As a result, no fires scenarios are run where the cabinet fires damage overhead cables, for either control	PARTIALLY RESOLVED – OPEN DOCUMENTATION Modeling of the CSR scenarios is documented in the Fire Scenario Development (FSD) notebook Appendix A, Part 2. The general description of the fire scenarios point to a correct approach for use of proper inputs, model and assumptions. The practice, as explained by the PBAPS Fire PRA engineer during the review is appropriate. However, it is not clear if the process used to select scenarios could lead to overly conservative results. Further resolution of this F&O has no impact on any technical element of the analysis. This is a documentation issue with no impact on this application.

Finding Number	Supporting Require-ment(s)	Capability Category (CC)	Finding Description	Disposition for 50.69
			room abandonment scenarios or non-abandonment scenarios. Additionally, non-suppression probabilities are not included, as well as other factors affecting fire growth, damage and timing.	
2012-1-16	FSS-D8	Not Met	For Cable Room Fires, the assumptions used to analyze the Fire Scenarios are includes conservative estimates for non-suppression: 1) Once Damage occurs, since actuation of the CO2 System would result in a control room abandonment, the ASD event tree logic is used for all fires damaging overhead cables - regardless of the cables damage and the extent of damage. No credit for manual suppression without actuation of CO2 is taken, other than the initial credit for non-suppression in the calculation of the probability to damage the first cable tray - performed in the MathCad Calculation. In discussion with the PB/corporate Fire Protection Engineer; the strategy	PARTIALLY RESOLVED – OPEN DOCUMENTATION The Fire Modeling Treatment (FMT) Notebook, Appendix E Section E.1, describes application of the non-suppression probability, manual and automatic. This section and the Fire Scenario Development (FSD) Notebook, Attachment A2, shows that manual suppression without actuation of CO2 is credited for cable spreading room scenarios. The FMT Notebook, Appendix E Section E.1, also shows how the manual CO2 system was credited in the event of the failure of the manual suppression system. Credit for various suppression systems requires consideration of the dependency between those systems. Based on conversations with the Fire PRA Engineer, this dependency was considered but is not documented. This is a documentation issue with no impact on this application.

Finding Number	Supporting Require-ment(s)	Capability Category (CC)	Finding Description	Disposition for 50.69
			for the cable spreading room was modified approximately 6 years ago, through NRC submittal, to include various strategies and equipment including telescoping nozzles, thermal imaging, etc. As a result, the use of CO2 is likely not needed, and is typically a last resort.	
2012-1-33	FSS-C4	Met CC I	The Transient Fires used in the Fire PRA include an area weighting factor based on the ZOI divided by the total area. A review of several weighting factors was performed. For example, scenario 06S_508_T11:B was reviewed. The ZOI assigned was based on a 3' by 1' ZOI, even though the scenario is in the corner. No basis for the factor was provided. Scenario 13N_523_T03:C is listed as 9.1' V x 8.4', although this appears to be not used and is lower (based on hand calculations of the severity factor). It appears many of the weighting factors are calculated based on the distance to the first target. However,	PARTIALLY RESOLVED There are two parts to this finding. 1) Application of floor-area-ratio (FAR) when the transient fire is located against the wall or in a corner leading to a smaller FAR. This issue is resolved based on use of Zone of Influence (ZOI) to define the FAR and the fact that the ZOI accounts for fires against a wall or in a corner. In other words, such fires lead to larger ZOI. 2) Calculation of the FAR in the transient fire scenario frequency. The PBAPS Fire PRA calculates the FAR based on the floor area derived from the ZOI of the target divided by the total floor area of the fire compartment. Use of this approach requires consideration of: a) obstructed floor space where placement of transient fire sources is unlikely, and b) consideration of maintenance practice. Maintenance

Finding Number	Supporting Require-ment(s)	Capability Category (CC)	Finding Description	Disposition for 50.69
			this distance is not provided in the FSS documentation. Additionally, there are areas within some areas (e.g., 06S) where transients are assumed to not be located. This area is not removed from the denominator of the area factor. Basis for the weighting factors is not provided. Given that more than 100 of the transient fire frequencies are below 1E-06/year, the overall impact of the weighting factors appears to result in lower than expected transient fire frequencies.	practice may provide indication that likelihood of placement of transient fire sources are more likely than others. To address this F&O, the floor area ratios (FARs) are being adjusted based on the configuration of the PAU to account for unequal distribution of transient frequency across the entire PAU. The use of a FAR is consistent with the guidance in NUREG/CR-6850 Section 11.3.1. The resulting adjustment to the transient fire scenario frequencies is being resolved in the current model update and has minimal impact on this application.
2012-1-40	FSS-C1	Met CC I	The current FRANX file contains numerous risk significant fire scenarios (Fussell-Vesely > 1%, or in the top 95% of CDF/LERF) scenarios that are modeled as bounding fire scenarios (single size, no growth, no damage time calculation, etc.) in the FPRA (02_236_F01:D, 38_217_F06:E, 36_226_F05:E, 36_226_F06:E, and others), where information for severity factor or non-suppression	PARTIALLY RESOLVED The top risk contributing scenarios are identified in the Summary and Quantification Report. Figure 4-5 and Figure 4-6 of that report shows only 1 scenario that does not include detailed fire modeling. A review of the examples provided in the F&O statement show that they now include detailed fire modeling. However, some fire scenarios that are risk significant have not been analyzed using this approach. The review team's recommendation was to apply the two-

Finding Number	Supporting Require-ment(s)	Capability Category (CC)	Finding Description	Disposition for 50.69
			factor (including credit for automatic detection and suppression) appears to be incomplete, or based on the distance to the nearest target only, resulting in unrealistically high CDF values.	point fire modeling for the remaining risk significant scenarios. Two-point fire modeling is being applied to risk significant scenarios as appropriate. This F&O is being resolved in the current model update and has minimal impact on this application.
2012-2-6	HRA-A3	Met CC I	The Fire HRA Notebook, PB-PRA-021-04, provides an explanation in Section 2.3 that is intended to justify why instruments that could potentially mislead the operators or cause them to perform a harmful action were not included in the Fire PRA. Because this is a general discussion based on operator interviews and not on an instrument-by-instrument or procedure-by-procedure review, this SR is met at CCI, which is 'No requirement.'	OPEN To address the F&O a procedure-by-procedure review is being performed. If justification cannot be provided the undesired operator actions will be incorporated into the PRA model when applicable. This F&O is being resolved in the current model update and has minimal impact on this application.
2012-2-7	ES-C2	Not Met	Instruments that indicate critical parameters needed by operators in executing the EOPs are included in the SSEL and the Fire PRA. These parameters are: CST Level, RPV Level, RPV Pressure, Drywell Temperature, Drywell Pressure, Torus Temperature and Torus Level. The Fire HRA	OPEN This F&O is related to Finding Number 2012-2-6. To address the F&O a review of the alarm response procedures is being performed. If justification to exclude the instrument cannot be provided, it is included in the equipment selection task.

Finding Number	Supporting Require-ment(s)	Capability Category (CC)	Finding Description	Disposition for 50.69
			Notebook, PBPRA- 021-04, provides an explanation in Section 2.3 that is intended to justify why instruments that could potentially mislead the operators or cause them to perform a harmful action were not included in the Fire PRA. Because this is a general discussion based on operator interviews and not on an instrument by- instrument review, this SR is judged as being Not Met.	This F&O is being resolved in the current model update and has minimal impact on this application.
2012-3-14	FSS-G4	Not Met	Peach Bottom barrier feature condition reports were not reviewed to identify if the barrier random failure probabilities for doors, dampers, penetration seals identified in NUREG/CR-6850 are appropriate for use in PB-PRA-021.06, Multi-Compartment Analysis Notebook.	PARTIALLY RESOLVED – OPEN DOCUMENTATION Appendix A of the Fire Modeling Treatment (FMT) notebook explains the site-specific barrier unavailability data is collected and used to demonstrate applicability of the generic barrier failure probabilities. The technical is resolved. However, the resolution needs to be properly documented. This is a documentation issue with no impact on this application.
2012-3-17	FSS-C8	Not Met	A technical basis for the fire resistance of the embedded cables, encapsulated raceways, and heat shields credited in the Fire PRA analysis is not provided. PB-PRA-021.05 does not include an	PARTIALLY RESOLVED The references that contains the technical basis for the fire resistance-rating are provided in Appendix B of the Fire Scenario Development notebook.

Finding Number	Supporting Require-ment(s)	Capability Category (CC)	Finding Description	Disposition for 50.69
			assessment of the fire wrapped raceways with respect to mechanical damage from a high energy arcing fault or from direct flame impingement.	In addition, the appendix addresses the possibility of mechanical damage. App B says "There were some locations that were identified as potentially vulnerable to mechanical damage; however, this has not been confirmed. Therefore, the fire PRA currently assumes the fire wrap is successful in protecting the enclosed cables." To address this F&O a closer review of the potentially vulnerable wrap configuration is being performed. If the wrap is confirmed to be susceptible to mechanical damage the applicable fire scenarios are being updated to not take credit for the fire wrap. The current data indicates that no credited fire wrap is susceptible to mechanical damage. Resolution of this F&O will have minimal impact on this application.
2012-3-18	FSS-D6	Met CC I/II/III	Peach Bottom has used a heat release rate of 317 kW for transient fires throughout the fire modeling, except for the 4kV Switchgear Building Corridor, which used a 69 kW fire size. There is no documentation provided to justify the use of this fire size and whether it is within the limitations and conditions of the EPRI method.	PARTIALLY RESOLVED To resolve this F&O the justifications for postulating lower HRRs in select PAUs is being updated to include more details on the PAU attributes and credited administrative controls. A review of violations of the transient combustible controls program is also being performed to determine the impact on the HRR justifications.

Finding Number	Supporting Require-ment(s)	Capability Category (CC)	Finding Description	Disposition for 50.69
				The FMT notebook provides justification that using smaller transient HRR, such as 60 kW or 145 kW, is appropriate in some rooms based on the types of transients expected. Application Specific Model Notebook PB-ASM-10 provides justification for each PAU using a transient HRR of 60 kW or 145 kW. However, the justification does not fully address the NRC guidance for using lower HRR values (NEI letter dated September 27, 2011, referenced in ADAMS ML12171A583). This F&O is being resolved in the current model update and has minimal impact on this application as the resolution is a documentation enhancement.
2012-3-22	FSS-D1	Not Met	The fire modeling tools selected for use are appropriate for evaluating the zone of influence associated with individual fixed and transient ignition sources. However, the use of the generic fire modeling treatments does not provide an adequate method for fire growth and damage behavior for fire scenarios involving ignition and fire spread on secondary combustibles. The detailed fire modeling performed in Appendix E of the Fire Scenario	PARTIALLY RESOLVED The Fire Modeling Treatment (FMT) notebook documents that the method has changed and fire growth, spread, and damage behavior are now included. Cable trays as secondary combustibles are modeled using FLASH-CAT. The 5-minute to ignition of cable tray is no longer used. One-hour fire duration is no longer used as a limit. The resolve this F&O the Fire Modeling Workbook (FMW) is being updated to document the consideration of propagation in the fire modeling calculations.

Finding Number	Supporting Require-ment(s)	Capability Category (CC)	Finding Description	Disposition for 50.69
			Notebook does not incorporate cable tray and any other secondary combustibles, or adjacent vertical sections in the fire growth analysis. Based on the incorporation of the additional combustible fire growth profile, the one-hour fire duration may be exceeded and the additional HRR would increase the subsequent zone of influence. The basis for ignition of a cable tray above the fire source at 5 minutes regardless of actual distance between the fire source and cable tray is not justified.	Resolution of the F&O has negligible impact on the 50.69 categorization process.
2012-3-34	FSS-A5	Not Met	Potential transient fire scenarios have not been included in the fire modeling for some PAUs. For example, open areas of the Cable Spreading Room with low horizontal cable trays above have not been considered for transient fire modeling. Transients appear to have only been modeled at pinch-points (i.e., vertical risers, very low trays, and around some significant panels). The basis for transients being bound by the fixed source frequency has not been quantitatively justified, nor has the	PARTIALLY RESOLVED – OPEN DOCUMENTATION Transient fire scenarios may be located at the pinch-point. The frequency of the ignition will need to be justified if a fraction of the total PAU transient fire frequency is used. The total of all locations must add to the total PAU transient frequency. Appendix F of the Ignition Frequency notebook shows appropriate accounting of frequency. This is a documentation issue with no impact on this application.

Finding Number	Supporting Require-ment(s)	Capability Category (CC)	Finding Description	Disposition for 50.69
			basis that enclosed equipment will not fail due to a transient fire adjacent to the equipment.	
2012-3-37	FSS-A5	Not Met	Not all of the modeled non-propagating electrical panel fires considered failure of the panel and all targets terminating at the panel. Justification is based on the panel failures being addressed in the internal events failure, however, no documentation of this is provided in PB-PRA- 021.05.	OPEN To resolve the F&O it is being confirmed that the excluded panels lead to a single failure and if it is confirmed multiple failures could occur, the panel will be included in the fire PRA. This F&O is being resolved in the current model update and has minimal impact on this application.
2012-3-39	FSS-A6	Met CC I/II	The Main Control Room Abandonment Calculation assumes 15 minute spread to the adjacent cabinet. However, there is no documentation that a visual inspection was performed to ensure that the cables are not located against the wall in the adjacent cabinet or whether there are cable runs between the panels.	OPEN To resolve this F&O the time for fire spread to the adjacent cabinet is being changed to 10 minutes as it was not confirmed that the conditions to use 15 minutes are satisfied. Main control abandonment scenarios due to habitability were not significant contributors in the fire PRA. This F&O is being resolved in the current model update and has minimal impact on this application.

Finding Number	Supporting Require-ment(s)	Capability Category (CC)	Finding Description	Disposition for 50.69
2012-4-11	ES-B2	Met CC I	A review of the MSO development revealed the following incompleteness for scenarios that impact system success criteria: None of the new MSO scenarios in NEI-00-01 Rev. 3 have been evaluated and dispositions or modeling provided. • Scenario 2b – It is not clear how the FPRA modeling address potential impacts to ECCS pump NPSH. (See 0.5 probability for MSO termination from FPIE). • Scenario 2e – This scenario is screened from the FPRA with no substantial basis provided. • Scenario 2i.2 – It is not clear that ability of containment vacuum breakers to accommodate spraying the drywell into a hot dry environment has been addressed • Scenario 2j – It is not clear that potential pump runnout, and the possibility that suppression pool cooling, in addition to LPCI, may be impacted. • Scenario 2l – 'Event Modeled?' indicates RHR flow diversion, but this	PARTIALLY RESOLVED Each MSO scenario and the initiating event potential were reviewed as follows: • First part of F&O notes that new MSO scenarios in NEI-00-01 Rev.3 are not addressed. These MSO scenarios were confirmed to have been added. • 2b. The specific concerns of this F&O are identical to the concern in F&O 4-18 item (F). Assessment of Scenario 2b is provided for F&O 4-18 and is tracked to final resolution under that F&O. The Reference/ Resolution provided here, however, raised concerns beyond the scope of the F&O regarding application to the FPRA of the deterministic criteria used in technical evaluation for Scenario 2b. Per the suggestion subsequent to the F&O review, it was verified that deterministic criteria was not used to exclude cables that should be included and addressed using Circuit Mode Failure Likelihood Analysis (CFMLA). • 2e. Table B-1 of ES NB says that scenario is not modeled in PRA, yet the response here indicates that it is. F&O 4-2 evaluated this also and it is screened (scenario cannot occur). F&O resolution

Finding Number	Supporting Require-ment(s)	Capability Category (CC)	Finding Description	Disposition for 50.69
			scenario addresses RHR operation without minimum flow (Documentation). • Scenario 2p – It is not clear that the plant modification to address this scenario has been implemented. • Scenario 2w and scenario 2z – These scenarios are not modeled, yet they potentially impact HPCI / RCIC. • Scenario 2y – Potential impacts to pump NPSH due to loss of suppression pool inventory have not been addressed. • Scenario 2ag – Only one suppression pool draindown pathway was considered. • Scenario 5a – Loading of additional components onto DG can lead to under-voltage and tripping of all loads, which could require local tripping of circuit breakers, diesel start, and manual loading of the desired DG loads. This potential has not been addressed. If it is found that this scenario is to be included as a result of the finding, success criteria will need to be developed: identify diesel responses to spurious loading	requires update. • 2i.2. Table B-1 of ES Notebook includes discussion to address F&O issue. • 2j. Table B-1 of ES Notebook includes discussion to address F&O issue. • 2l. Table B-1 of ES Notebook and Table F-1 of PRM Notebook updated to address F&O issue. • 2p. Table B-1 MOD Completed column, states "Open Breaker to MO-176." PEAF-0015, FSSD MSO Analysis, confirms MOD completion. Table F-1 of PRM Notebook does not address MOD as stated in Resolution. • 2w, 2z. Table B-1 of ES Notebook includes discussion to address F&O issue. These scenarios are not modeled so they are not listed in Table F-1 of PRM Notebook as stated in the Resolution. • 2y. Text in Table B-1 discusses additional failures required to result in low SP inventory which would result in not meeting required pump NPSH. Scenario deemed not credible because spurious operation of 4 MOVs would be required.

Finding Number	Supporting Require-ment(s)	Capability Category (CC)	Finding Description	Disposition for 50.69
			of single or multiple extra loads. • Scenario 5c – MO-2972 and MO-3972 are not mentioned as receiving a modification, but it is not clear that this modification has been implemented. • Scenario 5e – No substantial basis is provided to justify that this scenario does not impact the HPSW success criteria. • Scenarios 5f, 5g, 5h – These scenarios, with potential IE impacts are not modeled and no substantial FPRA basis is provided. • Scenario 5i – If a diesel operates for some duration without cooling water then it is typically failed and non-recoverable. It is not clear that the FPRA addresses this scenario, specifically for control room evacuation scenarios. • Scenario 5k – Potential for valve damage precluding manual operation of MOVs is considered, but not the potential for rupture of the valve pressure boundary, and address phenomenological impacts of any flooding.	• 2ag. This part of the F&O is identical to part of F&O 4-2. Scenario 2ag referred to a concern that other suppression pool draindown pathways need to be considered. The 2016 F&O Technical Review Team found that it was clear that from PEAF-0015, FSSD MSO Analysis, and ES Notebook Table B-1, that the scenario for diversion of suppression pool inventory to the main condenser is no longer possible. During the F&O closure review, plant drawings were reviewed to confirm that no new pathways could lead to suppression pool drain down. The ES Notebook will be revised to reflect this review. Assessment of Scenario 2ag is provided for F&O 4-2 and will be tracked to final resolution under that F&O. • 5a. From review of the referenced technical evaluation, the F&O concerns and MSO-5a have been adequately addressed. • 5c. Table B-1 of ES Notebook includes discussion to address F&O issue. • 5e. The F&O Response does not provide details to assess resolution of the F&O concern. Question ARR-9 was submitted and contains more detail related to model changes that include spurious opening of valves leading to flow diversion and failing

Finding Number	Supporting Require-ment(s)	Capability Category (CC)	Finding Description	Disposition for 50.69
				the respective HPSW loop. Technical concern of F&O is addressed but F&O response should be updated to include detail from ARR-9 response. • 5f, 5g, 5h. This part of the F&O is identical to part of F&O 4-2. Assessment of Scenarios 5f, 5g, and 5 h is provided for F&O 4-2 and will be tracked to final resolution under that F&O. • 5i. Table B-1 of ES Notebook includes discussion to address F&O issue. • 5k. For IN 92-18, the response to ARR-12 was based on discussions with engineers and justification of typical weak link analyses. Only one example was provided. This is not believed to be sufficient justification to resolve the F&O. The additional valve list provided in the Application Specific Model Notebook PB-ASM-10 does not include details for the process used to determine the scope of valves listed as not having pressure boundary integrity issues. The F&O is being resolved by following the recommendation of the F&O Review Team to update the documentation to include the process used to determine the scope of valves listed as not having pressure boundary integrity issues. The remaining aspects of this F&O are

Finding Number	Supporting Require-ment(s)	Capability Category (CC)	Finding Description	Disposition for 50.69
				documentation issues with no impact on this application.
2012-4-23	AS-A5	Not Met	The FPIE accident sequence modeling that was carried over to the FPRA was not reviewed for consistency with the fire abnormal procedures (fire area guides). Such a review would confirm that the accident sequence modeling is consistent with these procedures, or prompt modeling changes to conform to them. A finding is made.	PARTIALLY RESOLVED – OPEN DOCUMENTATION The detailed area fire guides were reviewed and documented in the reference document as stated. The explanation of how this was considered in the accident sequence (AS) analysis is not discussed. However, this was discussed with the PRA team during the review, and it was explained that the fire area guides only affect the fault tree under an existing top event in the event trees, not the structure of the event trees. A spot check "challenge" was held, and confirmed the conclusion. The only exception was for scenarios involving main control room abandonment, and a new accident sequence model was used for that, which considered the associated fire area guide. This is a documentation issue with no impact on this application.
2012-4-30	HRA-C1	Met CC I	The following risk-significant HEPs are candidates for detailed HRA or more documentation indicating that the actions are infeasible. One is a screening HFE. The others are assigned 1.0 values and it is not clear	PARTIALLY RESOLVED – OPEN DOCUMENATION Table 3-1 of the Fire HRA notebook (PB-PRA-021.09) summarizes the bases for why certain actions were considered infeasible. On initial review, there was a little lack of clarity regarding what each basis meant,

Finding Number	Supporting Require-ment(s)	Capability Category (CC)	Finding Description	Disposition for 50.69
			from the documentation whether they are feasible. CDF: DHUAP035DXI2-FRA 0 HEP 1.00E-01 FV 3.78E-02 AHUBTL-LDXI2-F OPERATOR FAILS TO VALVE IN N2 BOTTLES (LOCALLY) HEP 1.00E+00 FV 6.16E-02 (may not be feasible) VHU-2511LPI2-F OP FAILS TO OPERATE AO-2511 HEP 1.00E+00 FV 5.62E-03 (may not be feasible) LERF (same HFEs as for CDF): AHUBTL-LDXI2-F OPERATOR FAILS TO VALVE IN N2 BOTTLES (LOCALLY) HEP 1.00E+00 FV 3.00E-02 (may not be feasible) DHUAP035DXI2-FRA 0 HEP 1.00E-01 FV 1.91E-02	but in response to a question sent by the review team, this was clarified and the bases are reasonable – these actions fit the definition of infeasible. This is a documentation issue with no impact on this application.
2012-4-9	ES-B1	Met CC I	The PRA basic events are reviewed and screened in Appendix A of the model development notebook. A review of the basic events screened produced the following observations: a. The following events are screened from the FPRA equipment list with the disposition: 'Based on review of	PARTIALLY RESOLVED – OPEN DOCUMENTATION Each observation was reviewed as follows: a) Events are no longer screened in current Equipment Selection (ES) notebook. Table A-1 of the ES Notebook shows that these events are mapped in the Fire PRA. The Plant Response Model (PRM) Notebook Table B-1 discusses the basic events added

Finding Number	Supporting Require-ment(s)	Capability Category (CC)	Finding Description	Disposition for 50.69
			schematic with Chris Pragman, the following events cannot cause failure mode.' It is not clear what the basis for screening is: DMV--34AHOI2 MOTOR OPERATED VALVE 10-34A SPURIOUSLY OPENS (NC-FO), DMV--34BHOI2 MOTOR OPERATED VALVE 10-34B SPURIOUSLY OPENS (NC-FO) DMV--39AHOI2 MOTOR OPERATED VALVE 10-39A SPURIOUSLY OPENS (NC-FO), DMV--39BHOI2 MOTOR OPERATED VALVE 10-39B SPURIOUSLY OPENS (NC-FO) b. The following events are screened from the FPRA equipment list with the disposition: 'Per review of the component masking, it is determined that this SSE cannot fail the component modeled in this BE.' It is not clear what the basis for screening is: DPMAP035DSI2 MOTOR DRIVEN PUMP 2AP035 FAILS TO START DPMBP035DSI2 MOTOR DRIVEN PUMP 2BP035 FAILS TO START	to the Fire PRA and provides the hot short failure basic events for the basic events identified in this finding. b) The basic events DPMAP035DSI2 and DPMBP035DSI2 (FTS of the RHR pumps A and B, respectively) are not directly mapped in the Fire PRA as the mapping is addressed by another logically equivalent basic event which in this case is the RHR pump A and B FTR basic events (DPMAP035HRI2 and DPMBP035HRI2). These FTR basic events are directly mapped in the Fire PRA model as can be seen in Table A-1 of the ES Notebook. c) Events were confirmed to be re-assessed in ES Notebook Appendix A, however, three of the basic events (ELB-0251HOI2, ELB-0221HOI2, ELB-0121HOI2) have disposition code N1E and are not included in the Fire PRA as stated in Resolution. F&O concern is resolved. d) The ES Notebook Table A-1 provides the following disposition for the VAV-2082DNI2 basic event, "Fire can't cause the failure mode" rather than disposition stated in Resolution. Revise Resolution to reflect disposition in NB. e) Confirmed events are no longer screened.

Finding Number	Supporting Require-ment(s)	Capability Category (CC)	Finding Description	Disposition for 50.69
			c. Basic events assigned as N2 because they are associated with systems such as FW, IA, SW that have assumed routing (e.g., EBS1T4TAHWI2, EBS2T4TAHWI2, EHB-0110HOI2, EHB-0204HOI2, ELB-0121HOI2, ELB-0142HOI2, ELB-0212HOI2, ELB-0221HOI2, ELB-0251HOI2, ELB-0272HOI2, EMC1G4TAHWI2, EMC1T4TCHWI2, EMC2G4TAHWI2, EMC2T4TCHWI2, FLC-DCCXHWI2): a) Numerous basic events that impact systems with assumed routing were screened from the FPRA equipment list. Thus, the assumed routing is performed at ahigh (system) level. At a minimum it is recommended that these basic events be assigned as Y3 and the ass umed routing be performed at the basic event / functional state level. b) The equipment selection is iterative, and the systems with assumed routing were found to be risk significant by side calculation sensitivity evaluations performed by the FPRA team. Therefore events such as these require a further step to be identified	f) WPNAP004HRIE2 confirmed to be re-assessed in ES Notebook Appendix A, however, TPV80029DWIE2 has disposition code N3 (should be N1E per ARR-8) and is not mapped in Fire PRA as stated in Resolution. F&O concern is resolved. Resolution of the above items is a documentation issue with no impact on this application.

Finding Number	Supporting Require-ment(s)	Capability Category (CC)	Finding Description	Disposition for 50.69
			for cable selection (Y* designations other than Y3). d. The following events are screened from the FPRA equipment list with the disposition: 'This valve is designed to automatically open during an accident to allow drainage. However, if cables associated with valve are affected by a fire it would not be able to open. As a result the valve would not be required to reclose. ' It is not clear what the basis for screening is: VAV-2082DNI2 AO-20-82 FAILS TO CLOSE (FO) (and 3 similar events) e. [e. The following events are screened from the FPRA equipment list with the disposition: 'Special Initiator-Like Event' It is not clear what the basis for screening is: EBS00A03HWI0IEW, Bus Fails To Operate (and 8 similar events) EHB-0306HOI0IEW >4.16 kVAC Circuit Breaker Spuriously Opens (NC-FO) (and 15 similar events) EXH00X11HWI0IEW, TRANSFORMER 00X11 343 STARTUP FAILS LOSS OF	

Finding Number	Supporting Require-ment(s)	Capability Category (CC)	Finding Description	Disposition for 50.69
			FUNCTION (and 6 similar events) f. The following events are screened from the FPRA equipment list with the disposition: 'Support System Initiating Event Logic' It is not clear what the basis for screening is: TPV80029DWIE2 PRESSURE CONTROL VALVE 34-80029 WPNAP004HRIE2 MOTOR DRIVEN PUMP NSW 2AP004 FAILS TO RUN	
2012-5-1	CS-B1	Not Met	Did not 'IDENTIFY any additional circuits and cables whose failure could challenge power supply availability due to inadequate or unanalyzed electrical overcurrent protective device coordination.' Specific examples not identified or documented	OPEN To resolve this F&O the breakers for which an existing coordination calculation does not exist will either be treated as non-coordinated in the PRA model or have a coordination calculation or other industry approved method performed to confirm it is coordinated. This F&O is being resolved in the current model update and has minimal impact on this application.
2012-5-6	FSS-E3	Not Met	Individual fire modeling references generally provide qualitative uncertainty treatment and in some cases sensitivity studies. The individual fire scenarios are treated in the Fire Scenario Notebook (PBPRA-	OPEN To resolve this F&O the uncertainties of fire modeling parameters for the risk significant fire scenarios are being included in the Fire Modeling Treatments Notebook. Sensitivities are being provided in the

Finding Number	Supporting Require-ment(s)	Capability Category (CC)	Finding Description	Disposition for 50.69
			021.05). Statistical representations are generally not provided	Uncertainty and Sensitivities notebook to determine the impact of varying the parameters with a significant source of uncertainty. This F&O is being resolved in the current model update and has minimal impact on this application.
2012-6-1	PRM-B2	Not Met	Details of the FPIE Peer Review are contained in Appendix H of the Summary and Quantification Notebook (PB-PRA-021.01). Discussion of impact on Fire PRA for some 'not-met' supporting requirements lack sufficient detail for a reviewer to confirm that the impact is low. Examples include: 1) Finding 6-1 in Table H-1 involves use of maintenance rule unavailability data without validating it. Assessment of impact only states that values used are sufficient for providing an appropriate level of accuracy. 2) Finding 3-6 in Table H-2 involves review of plant test procedures to define pre-initiator activities that would cause plant trips. Assessment of impact for this open item only	PARTIALLY RESOLVED The Plant Response Model Notebook PB-PRA-021.55 Sections 4.1.4 and 4.1.5 summarize FPIE peer reviews and impact on Fire PRA. Appendix C of this same document addresses disposition of peer review findings and suggestions. 1) This should refer to FPIE Finding 6-11 (DA-C11), not 6-1 in PB-PRA-021.01 and the FPIE PR report – this item is still open. 2) FPIE Finding 3-6 (HR-C2, QU-D6; Note that this finding should have been written against HR-A1/A2) in PB-PRA-021.01 and the FPIE PR report – this item is still open (UREPB2011-29). The SR does not allow flexibility regarding performing the review of plant test procedures. 3) Finding 5-9 was considered resolved by the 2016 F&O Technical Review Team. Systems that support the reactivity control function were included in the

Finding Number	Supporting Require-ment(s)	Capability Category (CC)	Finding Description	Disposition for 50.69
			states that failure modes identified in the SR are already included in the generic or plant-specific data utilized for each system, component, and initiating events. 3) Finding 5-9 in Table H-2 involves level of development for systems that support the reactivity control function (i.e., RPS, ARI, RPT and SLC.) This open item appears to be assessed as having a low impact based only on a judgment that ATWS sequences would be below the truncation limit. 4) Supporting requirement SC-B5 was assigned not met for the FPIE peer review. The FPRA development did not resolve this FPIE assessment based on the following: No impact given that the current success criteria have been validated based on plant-specific MAAP runs or other comparable generic sources.	recirculation pump trip (RPT) and alternate rod insertion (ARI) in the standby liquid control (SLC) Notebook and model. They were also found in the PB214A model. Reactor protection system (RPS) modeling is black box similar to other plants. 4) This was related to Finding 6-5 (SC-B5) which was considered resolved by the 2016 F&O Technical Review Team. The cited document (Success Criteria Notebook PB-PRA-003, Section 3.5) provides a comparison between the Peach Bottom and Limerick success criteria. This F&O has not been resolved but has minimal impact on the 50.69 categorization process. This F&O was opened to track the open findings from the FPIE PRA Peer Review (see F&Os 6-11 and 3-6). These FPIE PRA findings were assessed as not being resolved but having minimal impact on the 50.69 process.
2016-1-1	FSS-D3	Not Met	THIEF Model: Section 2.2 of the FMT states that "It was assumed that control cables would be the critical target cable in most scenarios and	OPEN To resolve this F&O the assumption that control cables were the critical target is no longer made.

Finding Number	Supporting Require-ment(s)	Capability Category (CC)	Finding Description	Disposition for 50.69
			therefore these parameters were chosen." No basis is provided for this assumption to indicate that it is applicable to the risk-significant scenarios or that it is bounding. The same assumption acknowledges that this is a source of uncertainty. However, there is no indication in the UNC notebook that impact of this assumption was addressed.	Instead, weighted averages of the cable parameters located in the PAU are used for the THIEF calculations when they are not based on specific target cables. Sensitivities are also being performed to determine the impact of using representative upper bound and lower bound parameters instead of the weighted average. Parameters values that represent upper and lower bound values for approximately 90% of the cables are used in the sensitivities. This F&O is being resolved in the current model update and has minimal impact on this application.
2016-1-2	FSS-D4	Met CC I/II/III	THIEF Model: Page C-3 of the FMT notebook states that "The radial increment to be applied during the one dimensional heat transfer finite difference, specified in millimeters. A value of 0.25 mm was utilized." The FMW shows use of 0.5. This value is different than the recommended value in the NUREG/CR-6931, Volume 3. No basis is documented for use of a number outside the validated range.	OPEN To resolve this F&O a sensitivity study is being performed to show that the use of radial increment values larger than that recommended in NUREG/CR-6931, Volume 3 provides a reasonable approximation of the target delay time with some degree of conservatism but requires less processing time which enables a more efficient application of the model. This F&O is being resolved in the current model update and has minimal impact on this application.
2016-1-3	FSS-D6	Not Met	THIEF Model: NUREG/CR-6931 Volume 3, Section 6.0 described the limits of the THIEF model validation as	OPEN To resolve this F&O the use of the model within the

Finding Number	Supporting Require-ment(s)	Capability Category (CC)	Finding Description	Disposition for 50.69
			"The cables included in the study ranged from 7 mm (0.25 in) to 19 mm (0.75 in) in diameter, a common size for control cables, plus some instrument and low power cables. The copper content by volume ranged from 0.07 to 0.36 and the content by mass ranged from 0.31 to 0.89." No basis is documented that the model is used within its validation range.	range of parameters experimentally tested in NUREG/CR-6931 will be documented and a basis will be provided if used outside the range. This F&O is being resolved in the current model update and has minimal impact on this application.
2016-1-4	FSS-D2	Not Met	TIME TO AUTOMATIC DETECTION: The model has not been shown to be used within the known limits of applicability.	OPEN This F&O is being resolved in the current model update. As this is a documentation enhancement the resolution will have no impact on the categorization of SSCs.
2016-1-5	FSS-D8	Not Met	TIME TO AUTOMATIC DETECTION: The assessment of the effectiveness of the detectors in each scenario is not specifically addressed.	OPEN To address the F&O the effectiveness of the detectors is being documented. No impact to the application of the model has been identified and therefore this is only a documentation enhancement. The resolution of the F&O will have no impact on the 50.69 categorization process.
2016-1-6	FSS-H5	Met CC I	FLASH-CAT: Output results from parameter uncertainty evaluations are not documented. Some inputs, such	OPEN To resolve the F&O sensitivities are being performed

Finding Number	Supporting Require-ment(s)	Capability Category (CC)	Finding Description	Disposition for 50.69
			as Yp (mass fraction non-metallic cable material) and m_cable (mass per length) are uncertain and warrant sensitivity analyses. THIEF: Output results from parameter uncertainty evaluations for mass per length and initial radial increment are not documented.	in which the parameters are being varied to reasonable upper and lower bound approximations to determine the impact on the analysis. This F&O is being resolved in the current model update and has minimal impact on this application.
2016-1-7	FSS-H9	Not Met	THIEF: Sources of uncertainty are not documented. TIME TO AUTOMATIC DETECTION: Sources of uncertainty for the model are not addressed.	OPEN To resolve this F&O the sources of uncertainty for the time to automatic detection calculations will be documented. This F&O is being resolved in the current model update and has minimal impact on this application.
2016-1-8	FSS-H10	Met CC I/II/III	FLASH-CAT: Detailed walkdown notes supporting FLASH-CAT are not documented. TIME TO AUTOMATIC DETECTION: Walkdown notebook does not address site-specific conditions of the credited smoke detectors.	OPEN To resolve this F&O the walkdown notes to support FLASH-CAT and Time to Automatic Detection calculations will be documented. This F&O is being resolved in the current model update and has minimal impact on this application.
2016-1-9	FSS-D6	Not Met	The ERIN Fire Modeling Methodology, Verification of the ERIN Fire Modeling Workbook does not fully address verification of FLASHCAT and THIEF.	OPEN To resolve this F&O the verification of the implementation of FLASHCAT and THIEF in the fire

Finding Number	Supporting Require-ment(s)	Capability Category (CC)	Finding Description	Disposition for 50.69
				modeling workbook are being documented. This F&O is being resolved in the current model update and has minimal impact on this application.

Attachment 4: External Hazards Screening

External Hazard	Screening Result		
	Screened? (Y/N)	Screening Criterion (Note a)	Comment
Aircraft Impact	Y	C2 C3	None of the four local airports located near the plant site has scheduled commercial air service. The nearest major airports are Baltimore Washington International (BWI) and Philadelphia International (PHL) airports; each of which is located more than 40 miles from the site. There exist federally controlled air corridors that pass within one mile of the plant site; the IPEEE reports results from bounding assessments to demonstrate that the risk due to this hazard is less than 1E-6/yr. A review of Federal Air Administration data confirmed the IPEEE conclusion is still valid.
Avalanche	Y	C1 C3	Plant site not located near large mountains where snow avalanches are prevalent. Plant located in river valley at the bottom of a bluff. Worst case impact of cascade of snow or rock is damage to exterior structure of turbine and radwaste processing buildings.
Biological Event	Y	C5	Hazard is slow to develop and can be identified via monitoring and managed via standard maintenance process. Additionally, the hazard was

External Hazard	Screening Result		
	Screened? (Y/N)	Screening Criterion (Note a)	Comment
			assessed in the IPEEE where accumulation or deposition of vegetation or organisms and related blockage of the intake structure (which had previously resulted in a plant shutdown) was evaluated. The Emergency Cooling Tower provides an inherent plant design feature to provide an alternate source of cooling.
Coastal Erosion	Y	C3	PBAPS is a riverine site. The most severe occurrence of this hazard would be due to upstream dam failure which for precipitation-driven failure was screened.
Drought	Y	C5	Drought is a slowly developing hazard. The plant location (riverine site with upstream and downstream dams) preclude impact on plant due to this hazard.
External Flooding	Y	C1 PS2	The external flooding hazard at the site was recently updated as a result of the post-Fukushima 50.54(f) Request for Information and the Flood Hazard Reevaluation Report (FHRR) (Reference 14). The results indicate 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

External Hazard	Screening Result		
	Screened? (Y/N)	Screening Criterion (Note a)	Comment
			plant. Additionally, flooding from local intense precipitation was evaluated in the FHRR and determined to not challenge any safety functions at PBAPS.
Extreme Wind or Tornado	Y	PS2 PS4	Section 12.2 and Appendix C of the PBAPS UFSAR (Reference 15) describes the capability of safety related structures to withstand wind and tornado loadings. Structures that directly affect the ultimate safe shutdown of the plant are designed to resist applicable design basis tornado forces, which bound other winds at the Peach Bottom plant site. Most plant components which directly affect the safe shutdown of the plant are located either inside reinforced concrete structures or underground to provide protection against tornado winds and tornado generated missiles. The emergency cooling tower (ECT) structure/fans (Reference 16) and EDG exhaust stack (LER-2-17-001, Reference 17) can be damaged by tornado generated missiles; the frequency of damage to these components is estimated to be less than 1E-6/yr. Therefore, tornado missiles can be screened from

External Hazard	Screening Result		
	Screened? (Y/N)	Screening Criterion (Note a)	Comment
			consideration. The site is currently assessing tornado missile protection (TMP) in response to RIS 2015-06. If additional TMP vulnerabilities are discovered as part of this assessment, PBAPS will update this screening analysis.
Fog	Y	C1 C4	There is negligible impact on the plant for this hazard. The worst case impact is fog-induced freezing leading to a loss of off-site power event, which is an event evaluated in the PBAPS FPIE PRA.
Forest or Range Fire	Y	C4	The most significant consequence of a forest or range fire is a loss of off-site power (LOOP) which is evaluated in the FPIE model.
Frost	Y	C1 C4	There is negligible impact on the plant due to frost. The worst case impact is frost induced freezing leading to a loss of off-site power event which is evaluated in the PBAPS FPIE PRA.
Hail	Y	C1 C4	Hail is bounded by other events for which the plant is designed. Flooding impacts are covered under intense precipitation for external flooding. Additionally, in the IPEEE submittal, because PBAPS was designed to withstand 300 mph winds

External Hazard	Screening Result		
	Screened? (Y/N)	Screening Criterion (Note a)	Comment
			(design basis tornado), hail as a missile source was screened based on design basis protection requirements for tornadoes.
High Summer Temperature	Y	C1 C5	High summer temperatures are of negligible impact on the plant. This phenomenon provides large amount of time for preparation (weather forecast) with time for implementation of appropriate mitigation actions (e.g. plant power reduction or shutdown).
High Tide, Lake Level, or River Stage	Y	C1 C2	This event is of negligible impact on plant. The plant location (riverine with upstream and downstream dams) preclude impact on plant due to this hazard. Precipitation driven hydrologic dam failure was reevaluated in the Flood Hazard Reevaluation Report (Reference 14). The maximum resulting water surface elevation including wind-driven waves is below the current licensing basis (CLB) as reported in the PBAPS FSAR (Reference 15). The results from the FHRR indicate that flooding from rivers and streams and dam failure are bounded by the current licensing basis (CLB) and do not pose a challenge to the plant.

External Hazard	Screening Result		
	Screened? (Y/N)	Screening Criterion (Note a)	Comment
			See also External Flooding.
Hurricane	Y	C2 C4	The hurricane hazard was screened from evaluation in the FHRR (Reference 14). Hurricane winds are bounded by tornadoes.
Ice Cover	Y	C1 C2 C5	Plant structures have designed roof loads due to snow / ice of 30 psf (PBAPS UFSAR Appendix C "Structural Design Criteria" / Section C.2.1 "Dead and Live Loads" (Reference 15)). Additionally, the IPEEE noted that occurrence of frazzle ice resulted in plant shutdown during its operating history; however, a plant modification has been installed to prevent recurrence.
Industrial or Military Facility Accident	Y	C3	There are no military facilities within the proximity to the plant site (the closest is the Aberdeen Proving Grounds, MD, ~25 miles distant). The only large industrial facility within proximity to PBAPS is the Calpine York Generating complex located 3 miles from PBAPS. The only hazards from this facility are (1) the impact of toxic gas release from the facility and (2) the impact of a natural gas explosion at the facility. These specific hazards from this facility are addressed and were screened in the Pipeline Accident and Toxic Gas sections of this

External Hazard	Screening Result		
	Screened? (Y/N)	Screening Criterion (Note a)	Comment
			Table.
Internal Flooding	N	None	The PBAPS Internal Events PRA includes evaluation of risk from internal flooding events.
Internal Fire	N	None	The PBAPS internal Fire PRA addresses risk from internal fire events.
Landslide	Y	C4	Plant located in river valley at the bottom of a bluff. Per the IPEEE, detailed soil stability analyses were performed during site preparation to confirm the safety of the slopes on the north and west sides of the plant. The most likely impact of landslide from this bluff is damage to exterior structure of turbine and radwaste processing buildings. Worst case impact is plant trip with partial loss of off-site power event which is evaluated in the PBAPS FPIE PRA.
Lightning	Y	C2 C4	Lightning strikes may result in loss of offsite power or plant trip. These events are addressed in the plant design basis and are modeled in the PBAPS FPIE PRA. The IPEEE also indicated that plant operational experience indicated some plant radiation release monitoring instrumentation at the main stack has been affected by lightening; however these instruments do not represent a significant contribution to plant

External Hazard	Screening Result		
	Screened? (Y/N)	Screening Criterion (Note a)	Comment
			risk.
Low Lake Level or River Stage	Y	C3 C5	This hazard is of negligible impact on the plant. The plant location (riverine with upstream and downstream dams) precludes impact on the plant due to this hazard. The phenomenon also provides large amount of time for preparation (weather forecast) with time for implementation of appropriate mitigation actions (e.g. plant power reduction). Additionally, the IPEEE noted that telephone and radio links are maintained between Conowingo Dam and the plant and that upon detection of a significant uncontrolled release, the reactors will be shut down and the sluice gates closed with cooling switch-over to the emergency cooling tower and reservoir.
Low Winter Temperature	Y	C1 C5	This hazard is of negligible impact to the site. The phenomenon provides large amount of time for preparation (weather forecast) with time for implementation of appropriate mitigation actions (e.g. plant

External Hazard	Screening Result		
	Screened? (Y/N)	Screening Criterion (Note a)	Comment
			power reduction or shutdown).
Meteorite or Satellite Impact	Y	PS4	This hazard is of negligible likelihood of impact to the site (very low event frequency). This hazard also was reviewed as part of the IPEEE submittal and screened based on low frequency of occurrence.
Pipeline Accident	Y	C2 C5	IPEEE review indicated no chemical, oil, or gas pipelines are located within 5 miles of plant site. However, due to the construction and operation of the Calpine gas fired generation plant (located approximately 3 miles from the PBAPS site) this hazard cannot be screened via simple a distance screening criterion. A risk-informed evaluation was performed using NRC Regulatory Guide 1.91 (Reference 18). Included in the evaluation was a sensitivity evaluation to determine the time required to reach the amount of natural gas required to achieve a 1 psi peak pressure at PBAPS upon detonation. This sensitivity evaluation varied the pipeline capacity to release up to 3 times the gas flowrate for full power operation of all units. At a mass flow of 3 times the base case, the time required to achieve the

External Hazard	Screening Result		
	Screened? (Y/N)	Screening Criterion (Note a)	Comment
			required amount of gas is slightly greater than 1 day assuming no credit for detection or isolation. In addition, as indicated under discussion of the high winds hazard, critical plant structures are designed to withstand a tornado induced differential pressure of 3 psi. Since this is a factor of 3 greater than the 1 psi peak pressure wave experienced by the design basis explosion specified in RG 1.91, there exists substantial margin to protect critical plant SSCs located within Category 1 structures in the extremely unlikely event that such an explosion were to occur.
Release of Chemicals in Onsite Storage	Y	C4 PS1 PS2	The impact of releases of hazardous materials stored on-site was evaluated in the IPEEE submittal. Of the chemicals stored on site only carbon dioxide, nitrogen, and sulfuric acid required specific evaluation for impact on control room habitability. Based on the characteristics, volume, and location of these hazards, each was determined to be very unlikely to impact habitability of the main control room. Additionally, a Control Room

External Hazard	Screening Result		
	Screened? (Y/N)	Screening Criterion (Note a)	Comment
			Envelope (CRE) Habitability Program is required by plant Technical Specification 5.5.13; thus these are periodically reevaluated in accordance with the specification.
River Diversion	Y	C4	River level and flow are regulated by upstream and downstream dams. Water supply is not dependent upon inflow from local tributaries.
Sand or Dust Storm	Y	C1 C3	Plant site not located near sand dunes or other large source of small airborne particles.
Seiche	Y	C1 C2	The maximum water surface elevation (WSE) from a seiche was reevaluated in the FHRR (Reference 14). This flood causing mechanism was considered bounded by the design basis of the plant and the precipitation-driven hydrologic dam failure mechanism. The maximum WSE calculated from a seiche is below the 134.87 ft. plant protection level. There are no postulated adverse effect to safety-related SSCs required for safe shutdown from this mechanism.
Seismic Activity	N	None	See information in Section 3.2.3 of this application.
Snow	Y	C1 C5	This hazard is slow to develop and can be identified via monitoring and managed via normal plant processes.

External Hazard	Screening Result		
	Screened? (Y/N)	Screening Criterion (Note a)	Comment
Soil Shrink-Swell Consolidation	Y	C1 C5	This hazard is slow to develop and can be identified via monitoring and managed via normal plant processes.
Storm Surge	Y	C1 C2 C3	Flooding from the Probable Maximum Storm Surge (PMSS) was reevaluated in the FHRR (Reference 14). This flood causing mechanism was considered bounded by the design basis of the plant and the precipitation-driven hydrologic dam failure mechanism. There are no postulated adverse effect to safety-related SSCs required for safe shutdown from this mechanism.
Toxic Gas	Y	C4	Toxic gas is covered under release of chemicals in onsite storage, industrial or military facility accident, and transportation accident. Additionally, a Control Room Envelope (CRE) Habitability Program is required by plant Technical Specification 5.5.13. As part of this program hazardous materials stored at off-site facilities located near PBAPS are periodically evaluated in accordance with the specification. The analysis for chemicals stored at the Calpine Site is conducted on a periodicity of 6 years (Reference 19). The

External Hazard	Screening Result		
	Screened? (Y/N)	Screening Criterion (Note a)	Comment
			most recent assessment is documented in (Reference 20).
Transportation Accident	Y	C3	The impact of transportation accidents was evaluated in the IPEEE - Land Transportation: No interstate highways pass within 5 miles of the plant site. Traffic on local roads consists primarily of passenger and recreational vehicles and light weight trucks. Rail Transportation: The nearest active commercial rail traffic line is located across the Susquehanna River at a distance of 1.75 miles from the site. Additionally the IPEEE conducted an evaluation to demonstrate that the probability of a rail accident that resulted in release of toxic materials that could affect the site was less than 1E-6 /yr. Water Transportation: Conowingo Pond is not used for commercial water transportation.
Tsunami	Y	C1 C2 C3	Tsunami was not included in the FHRR as the mechanism was screened due to plant location.
Turbine-Generated Missiles	Y	C2 PS4	Turbine generated missiles are specifically analyzed and described in the plant UFSAR (Appendix C "Structural Design Criteria" / Section C.2.5.1 "Turbine Missiles") (Reference

External Hazard	Screening Result		
	Screened? (Y/N)	Screening Criterion (Note a)	Comment
			15). The turbine missile probability is less than 1E-5 per year, and the probability of damaging a critical target is maintained less than 1E-7 per year. These results are consistent with Sections 3.5.1.3 and 2.2.3 of the Standard Review Plan.
Volcanic Activity	Y	C3	Not applicable to the site because of location (no active or dormant volcanoes located near plant site).
Waves	Y	C1 C2	Waves were not considered as a separate hazard during the FPHH evaluation. They are bounded by other hazards that are considered and screen out (e.g., seiche).
Note a – See Attachment 5 for descriptions of the screening criteria.			

Attachment 5: Progressive Screening Approach for Addressing External Hazards

Event Analysis	Criterion	Source	Comments
Initial Preliminary Screening	C1. Event damage potential is < events for which plant is designed.	NUREG/CR-2300 and ASME/ANS Standard RA-Sa-2009	
	C2. Event has lower mean frequency and no worse consequences than other events analyzed.	NUREG/CR-2300 and ASME/ANS Standard RA-Sa-2009	
	C3. Event cannot occur close enough to the plant to affect it.	NUREG/CR-2300 and ASME/ANS Standard RA-Sa-2009	
	C4. Event is included in the definition of another event.	NUREG/CR-2300 and ASME/ANS Standard RA-Sa-2009	Not used to screen. Used only to include within another event.
	C5. Event develops slowly, allowing adequate time to eliminate or mitigate the threat.	ASME/ANS Standard RA-Sa-2009	
Progressive Screening	PS1. Design basis hazard cannot cause a core damage accident.	ASME/ANS Standard RA-Sa-2009	
	PS2. Design basis for the event meets the criteria in the NRC 1975 Standard Review Plan (SRP).	NUREG-1407 and ASME/ANS Standard RA-Sa-2009	
	PS3. Design basis event mean frequency is < 1E-5/y and the mean conditional core damage probability is < 0.1.	NUREG-1407 as modified in ASME/ANS Standard RA-Sa-2009	
	PS4. Bounding mean CDF is < 1E-6/y.	NUREG-1407 and ASME/ANS Standard RA-Sa-2009	

Event Analysis	Criterion	Source	Comments
Detailed PRA	Screening not successful. PRA needs to meet requirements in the ASME/ANS PRA Standard.	NUREG-1407 and ASME/ANS Standard RA-Sa-2009	

Attachment 6: Disposition of Key Assumptions/Sources of Uncertainty

Internal Events / Internal Flooding PRA Assumptions / Sources of Uncertainty		
Assumption / Uncertainty	**50.69 Impact**	**Model Sensitivity and Disposition**
The Loss of Offsite Power (LOOP) frequency and fail to recover offsite power probabilities are based on available industry data.	SSCs that support LOOP scenarios	The overall approach for the LOOP frequency and fail to recover probabilities utilized is consistent with industry practice and are representative of PBAPS. Therefore, this does not represent a key source of uncertainty and will not be an issue for 50.69 calculations.
Continued injection from control rod drive (CRD) after containment failure is credited unless a gross rupture of containment (i.e., not leak before break) occurs. The probability of rupture is based on a detailed structural analysis of the Mark I design.	SSCs that support containment heat removal scenarios	This approach provides a best estimate assessment for the site. Therefore, this does not represent a key source of uncertainty and will not be an issue for 50.69 calculations.
The base PRA model includes an assumption that the time during which the outside air temperature exceeds the design basis temperature for requiring two DG fans (> 80° F) is low for Peach Bottom and overheating of the diesels for the time frames required in the PRA is likely to be avoided even when only one fan is available. Therefore, only one fan is assumed to be required at all times.	SSCs supporting scenarios in which on-site AC power is required	This approach provides a best estimate assessment for the site. Therefore, this does not represent a key source of uncertainty and will not be an issue for 50.69 calculations.

Internal Events / Internal Flooding PRA Assumptions / Sources of Uncertainty		
Assumption / Uncertainty	**50.69 Impact**	**Model Sensitivity and Disposition**
The postulated reactor pressure vessel (RPV) overpressure failure mode is assumed to be equivalent to the Large LOCA success criteria.	SSCs supporting the LPI function in RPV overpressure failure LOCA scenarios	An alternative assumption would be that such scenarios are beyond the capabilities of the LPI systems. Therefore, crediting LPI capabilities for these scenarios may provide a slight non-conservative bias on the 50.69 calculations. However, because RPV overpressure LOCA scenarios are very low frequency events, this does not represent a key source of uncertainty and will not be an issue for 50.69 calculations.
The pipe rupture frequencies in the internal flooding PRA are based on an older version of the EPRI pipe rupture frequencies. Conversion to the most recent EPRI pipe rupture frequencies may increase internal flood CDF. The internal flood model uses a pipe length approach per EPRI TR-1013141 (Reference 21). Newer data is available.	SSCs that support Internal Flood scenarios	Updated industry data is developed routinely where it is common practice to implement this new data into the model during the next scheduled PRA Update. Therefore the PBAPS PRA model will incorporate the new pipe rupture frequencies during the next scheduled PRA model update. Any systems that have been analyzed in the 50.69 program prior to the scheduled model update will be re-evaluated after completion of the 2018 PRA model. Therefore, this does not represent a key source of uncertainty and will not be an issue for 50.69 calculations.

Internal Events / Internal Flooding PRA Assumptions / Sources of Uncertainty		
Assumption / Uncertainty	**50.69 Impact**	**Model Sensitivity and Disposition**
Credit for core melt arrest in-vessel at high RPV pressure conditions is taken in the current PRA model, but with a nominal failure probability of 0.9.	SSCs that support LERF scenarios	Core melt arrest in-vessel at high pressure may not be possible and therefore this could be a source of model uncertainty. Use of the 0.9 factor compared to the alternative assumption of 1.0 would not have a meaningful impact on the 50.69 calculations. However, prior to implementation of the 50.69 program, the PRA model will be updated to change this value to 1.0, such that this does not represent a key source of uncertainty for the 50.69 application.

Internal Events / Internal Flooding PRA Assumptions / Sources of Uncertainty		
Assumption / Uncertainty	**50.69 Impact**	**Model Sensitivity and Disposition**
Timely low pressure emergency core cooling system (ECCS) restoration after core damage is assumed to lead to a condition where vessel failure is avoided.	SSCs that support LERF scenarios	This assumption precludes some of the low likelihood phenomenological contributors to LERF from contributing to the overall results. However, it is judged reasonable that the availability of low pressure injection at the time of vessel failure (should it occur) will also greatly reduce the potential for a large early release from occurring. Therefore, this assumption provides a reasonable best-estimate approach, and as such will have only a minor impact on the 50.69 calculations. Therefore, this does not represent a key source of uncertainty for the 50.69 application.
Ex-vessel core melt progression overwhelming vapor suppression is considered in the LERF model with different values for low pressure RPV failure sequences and high pressure RPV failure sequences based on available information.	SSCs that support LERF scenarios	The values utilized provide a reasonable best-estimate approach, and as such will have only a minor impact on the 50.69 calculations. Therefore, this does not represent a key source of uncertainty for the 50.69 application.

Internal Events / Internal Flooding PRA Assumptions / Sources of Uncertainty		
Assumption / Uncertainty	**50.69 Impact**	**Model Sensitivity and Disposition**
The detailed Interfacing System LOCA (ISLOCA) analysis includes the relevant considerations listed in IE-C14 of the ASME/ANS PRA Standard and accounts for common cause failures and captures likelihood of different piping failure modes.	SSCs that support LERF scenarios	The values utilized provide a reasonable best-estimate approach, and as such will have only a minor impact on the 50.69 calculations. Therefore, this does not represent a key source of uncertainty for the 50.69 application.
Given the conditions occur that would allow uncontrolled flooding of the steam lines, a probability is assigned that this uncontrolled flooding permanently disables all of the SRVs precluding the ability to depressurize the RPV through the SRVs.	SSCs that support scenarios that require High Pressure Injection	Although the SRVs at PBAPS are designed to pass water and Appendix R models the RPV being flooded with water returning to the Suppression Pool via the SRVs, they are never tested in this fashion. A nominal failure probability is assigned to provide a slight non-conservative bias slant to the results such that the impact on 50.69 calculations is not unduly influenced. This does not represent a key source of uncertainty for the 50.69 application

Internal Events / Internal Flooding PRA Assumptions / Sources of Uncertainty		
Assumption / Uncertainty	**50.69 Impact**	**Model Sensitivity and Disposition**
Given the conditions occur that would allow uncontrolled flooding of the steam lines, a probability is assigned that this uncontrolled flooding permanently disables the HPCI or RCIC turbines.	SSCs that support scenarios that require High Pressure Injection	A steam turbine similar to RCIC has been tested by the manufacturer and successfully ran with a slug of water in the turbine. The RCIC and HPCI turbines are expected to behave in the same fashion due to similarity of design. A nominal failure probability is assigned to provide a slight conservative bias slant to the results such that the impact on 50.69 calculations is not unduly influenced. This does not represent a key source of uncertainty for the 50.69 application.

Internal Events / Internal Flooding PRA Assumptions / Sources of Uncertainty		
Assumption / Uncertainty	**50.69 Impact**	**Model Sensitivity and Disposition**
The probability that the suppression pool inventory could become ineffective as an inventory source due to the loss of NPSH during a large LOCA, ATWS, or in a transient with failure of suppression pool cooling is a potential source of uncertainty.	SSCs that support scenarios that require ECCS systems	There is a possibility that the lack of containment overpressure could lead to failure of the pumps in the scenarios of interest. As part of the EPU modifications, PBAPS implemented an RHR cross-tie mitigation strategy for the loss of NPSH scenario. Therefore a loss of NPSH would only result if the action to utilize the RHR cross-tie fails. Additionally, based on available information regarding pump NPSH requirements, it is concluded that pump operability would not be definitively impacted solely by the loss of containment overpressure arising from containment isolation failures since the pool conditions will not always be at the limited temperature assumed in the design basis calculations. A nominal failure probability is assigned to provide a slight conservative bias slant to the results such that the impact on 50.69 calculations is not unduly influenced. This does not represent a key source of uncertainty for the 50.69 application.

Internal Events / Internal Flooding PRA Assumptions / Sources of Uncertainty		
Assumption / Uncertainty	**50.69 Impact**	**Model Sensitivity and Disposition**
Containment integrity following a vessel rupture event (i.e., excessive LOCA) is not assured. There is model uncertainty regarding the subsequent treatment that increases the likelihood of LERF for this extremely rare event.	SSCs that support excessive LOCA scenarios	The current model treatment results in addition of a constant adder to the CDF and LERF results and as such will have only a minor impact on the 50.69 calculations. Therefore, this does not represent a key source of uncertainty for the 50.69 application.
There are model uncertainties associated with the likelihood associated with the extremely rare event of low intake pond level.	SSCs supporting systems that have dependencies on the intake pond as a source of water	The loss of pond event value is derived from expecting that an unrecoverable low intake pond level event would be an unlikely occurrence and could be averted based on reducing power levels prior to a plant trip, or by tripping the circulating water pumps following a plant trip if they had not already done so. Nominal failure probabilities are assigned to derive the overall likelihood that the precursor events (based on plant specific experience) proceed to a totally unrecoverable type of event. This method of implementation provides a slight conservative bias slant to the results such that the impact on 50.69 calculations is not unduly influenced. This does not represent a key source of uncertainty for the 50.69 application.

Internal Events / Internal Flooding PRA Assumptions / Sources of Uncertainty		
Assumption / Uncertainty	**50.69 Impact**	**Model Sensitivity and Disposition**
Detailed evaluations of HEPs are performed for the risk significant human failure events (HFEs) using industry consensus methods. Mean values are used for the modeled HEPs. Uncertainty associated with the mean values can have an impact on CDF and LERF results.	SSCs that use an Operator Action basic event as a surrogate for a modeled component	Sensitivity cases performed using the base internal events PRA (HEP values of 0.0 or use of the 95th percentile value HEPs) indicate some sensitivity to human performance. Use of 95th percentile HEPs for applications is not considered realistic given the consistent use of a consensus HRA approach. The PBAPS PRA model is based on industry consensus modeling approaches for its HEP calculations, so this is not considered a significant source of epistemic uncertainty. However, 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. These results are capable of driving a component and respective functions HSS and therefore the uncertainty of the PRA modeled HEPs are accounted for in the 50.69 application.

Internal Events / Internal Flooding PRA Assumptions / Sources of Uncertainty		
Assumption / Uncertainty	**50.69 Impact**	**Model Sensitivity and Disposition**
Dependent HEP values are developed for significant combinations of HEPs that have been demonstrated to appear together in the same cutsets.	Potentially all SSCs evaluated during 50.69 categorization	The PBAPS PRA model is based on industry consensus modeling approaches for its dependent HEP identification and calculations, so this is not considered a significant source of epistemic uncertainty. However, 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. These results are capable of driving a component and respective functions HSS and therefore the uncertainty of the PRA modeled HEPs are accounted for in the 50.69 application.

Internal Events / Internal Flooding PRA Assumptions / Sources of Uncertainty		
Assumption / Uncertainty	**50.69 Impact**	**Model Sensitivity and Disposition**
Common cause failure values are developed using available industry data.	Potentially all SSCs evaluated during 50.69 categorization	The PBAPS PRA model is based on industry consensus modeling approaches for its common cause identification and value determination, so this is not considered a significant source of epistemic uncertainty. Therefore, this does not represent a key source of uncertainty and will not be an issue for 50.69 calculations. Additionally, as directed by NEI 00-04, internal events common cause 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. These results are capable of driving a component and respective functions HSS and therefore the uncertainty of the PRA modeled CCF probabilities are accounted for in the 50.69 application.

Internal Events / Internal Flooding PRA Assumptions / Sources of Uncertainty		
Assumption / Uncertainty	**50.69 Impact**	**Model Sensitivity and Disposition**
There are model uncertainties associated with modeling the probability of the RHR pumps failing from a rupture due to a water hammer event given the RHR system is operating in suppression pool cooling mode at the time of the initiating event and the appropriate operator responses do not occur such that a potential water hammer event can occur.	SSCs supporting scenarios requiring RHR or RHRSW systems	The water hammer basic events and values utilized provide a reasonable best-estimate approach and will have only a minor impact on the 50.69 calculations. This does not represent a key source of uncertainty for the 50.69 application.

The table below describes the fire PRA sources of model uncertainty and their impact.

Fire PRA Sources of Model Uncertainty		
Description	**Sources of Uncertainty**	**Disposition**
Analysis boundary and partitioning	This task establishes the overall spatial scope of the analysis and provides a framework for organizing the data for the analysis. The partitioning features credited are required to satisfy established industry standards.	Based on the discussion of sources of uncertainly it is concluded that the methodology for the Analysis Boundary and Partitioning task does not introduce any epistemic uncertainties that would require sensitivity treatment. Therefore, the 50.69 calculations are not impacted.
Component Selection	This task involves the selection of components to be treated in the analysis in the context of initiating events and mitigation. The potential sources of uncertainty include those inherent in the internal events PRA model as that model provides the foundation for the FPRA.	In the context of the FPRA, the uncertainty that is unique to the analysis is related to initiating event identification. However, that impact is minimized through use of the BWROG Generic MSO list and the process used to identify and assess potential MSOs. Based on the discussion of sources of uncertainty and the discussion above, it is concluded that the methodology for the Component Selection task does not introduce any epistemic uncertainties that would require sensitivity treatment. Therefore, the 50.69 calculations are not impacted.

Fire PRA Sources of Model Uncertainty		
Description	**Sources of Uncertainty**	**Disposition**
Cable Selection	The selection of cables to be considered in the analysis is identified using industry guidance documents. The overall process is essentially the same as that used to perform the analyses to demonstrate compliance with 10 CFR 50.48.	Based on the discussion of sources of uncertainty it is concluded that the methodology for the Cable Selection task does not introduce any epistemic uncertainties that would require sensitivity treatment. Therefore, the 50.69 calculations are not impacted.
Qualitative Screening	Qualitative screening was performed; however, some structures (locations) were eliminated from the global analysis boundary and ignition sources deemed to have no impact on the FPRA (based on industry guidance and criteria) were excluded from the quantification based on qualitative screening criteria. The only criterion subject to uncertainty is the potential for plant trip. However, such locations would not contain any features (equipment or cables identified in the prior two tasks) and consequently are expected to have a low risk contribution.	In the event a structure (location) which could result in a plant trip was incorrectly excluded, its contribution to CDF would be small (with a CCDP commensurate with base risk). Such a location would have a negligible risk contribution to the overall FPRA. Based on the discussion of sources of uncertainty and the discussion above, it is concluded that the methodology for the Qualitative Screening task does not introduce any epistemic uncertainties that would require sensitivity treatment. Therefore, the 50.69 calculations are not impacted.

Fire PRA Sources of Model Uncertainty		
Description	**Sources of Uncertainty**	**Disposition**
Fire-Induced Risk Model	The internal events PRA model was updated to add fire specific initiating event structure as well as additional system logic. The methodology used is consistent with that used for the internal events PRA model development as was subjected to industry Peer Review. The developed model is applied in such a fashion that all postulated fires are assumed to generate a plant trip. This represents a source of uncertainty, as it is not necessarily clear that fires would result in a trip. In the event the fire results in damage to cables and/or equipment identified in Task 2, the PRA model includes structure to translate them into the appropriate induced initiator.	The identified source of uncertainty could result in the over-estimation of fire risk. In general, the FPRA development process would have reviewed significant fire initiating events and performed supplemental assessments to address this possible source of uncertainty. Based on the discussion of sources of uncertainty and the discussion above, it is concluded that the methodology for the Fire-Induced Risk Model task does not introduce any epistemic uncertainties that would require sensitivity treatment. Therefore, the 50.69 calculations are not impacted.

Fire PRA Sources of Model Uncertainty		
Description	**Sources of Uncertainty**	**Disposition**
Fire Ignition Frequency	Fire ignition frequency is an area with inherent uncertainty. Part of this uncertainty arises due to the counting and related partitioning methodology. However, the resulting frequency is not particularly sensitive to changes in ignition source counts. The primary source of uncertainty for this task is associated with the industry generic frequency values used for the FPRA. This is because there is no specific treatment for variability among plants along with some significant conservatism in defining the frequencies, and their associated heat release rates. PBAPS uses the ignition frequencies in NUREG-2169 (Reference 22) along with the revised heat release rates from NUREG-2178 (Reference 23).	Based on the discussion of sources of uncertainty, it is concluded that the methodology for the Fire Ignition Frequency task does not introduce any epistemic uncertainties that would require sensitivity treatment. Consensus approaches are employed in the model. Therefore, the 50.69 calculations are not impacted.
Quantitative Screening	Other than screening out potentially risk significant scenarios (ignition sources), this task is not a source of uncertainty.	The PBAPS FPRA did not screen out any fire scenarios based on low CDF/LERF contribution. That is, quantified fire scenarios results are retained in the cumulative CDF/LERF. Based on the discussion of sources of uncertainty and the discussion above, it is concluded that the methodology for the Quantitative Screening task does not introduce any epistemic uncertainties that would require sensitivity treatment. Therefore, the 50.69 calculations are not impacted.

Fire PRA Sources of Model Uncertainty		
Description	**Sources of Uncertainty**	**Disposition**
Scoping Fire Modeling	The framework of NUREG/CR-6850 includes two tasks related to fire scenario development. These two tasks are 8 and 11. The discussion of uncertainty for both tasks is provided in the discussion for Task 11.	Consensus modeling approach is used for the Detailed Fire Modeling. It is concluded that the methodology for the Detailed Fire Modeling task does not introduce any epistemic uncertainties that would require sensitivity treatment. Therefore, the 50.69 calculations are not impacted.
Detailed Circuit Failure Analysis	The circuit analysis is performed using standard electrical engineering principles. However, the behavior of electrical insulation properties and the response of electrical circuits to fire induced failures is a potential source of uncertainty. This uncertainty is associated with the dynamics of fire and the inability to ascertain the relative timing of circuit failures. The analysis methodology assumes failures would occur in the worst possible configuration, or if multiple circuits are involved, at whatever relative timing is required to cause a bounding worst-case outcome. This results in a skewing of the risk estimates such that they are over-estimated.	Circuit analysis was performed as part of the deterministic post fire safe shutdown analysis. Refinements in the application of the circuit analysis results to the FPRA were performed on a case-by-case basis where the scenario risk quantification was large enough to warrant further detailed analysis. Hot short probabilities and hot short duration probabilities as defined in NUREG 7150, Volume 2 (Reference 24), based on actual fire test data, were used in the PBAPS Fire PRA. The uncertainty (conservatism) which may remain in the FPRA is associated with scenarios that do not contribute significantly to the overall fire risk. Based on the discussion of sources of uncertainty and the discussion above, it is concluded that the methodology for the Detailed Circuit Failure Analysis task does not introduce any epistemic uncertainties that would require sensitivity treatment. Therefore, the 50.69 calculations are not impacted.

Fire PRA Sources of Model Uncertainty		
Description	**Sources of Uncertainty**	**Disposition**
Circuit Failure Mode Likelihood Analysis	One of the failure modes for a circuit (cable) given fire induced failure is a hot short. A conditional probability and a hot short duration probability are assigned using industry guidance published in NUREG-7150, Volume 2 (Reference 24). The uncertainty values specified in NUREG-7150, Volume 2 are based on fire test data.	The use of hot short failure probability and duration probability is based on fire test data and associated consensus methodology published in NUREG 7150, Volume 2. Based on the discussion of sources of uncertainty and the discussion above, it is concluded that the methodology for the Circuit Failure Mode Likelihood Analysis task does not introduce any epistemic uncertainties that would require sensitivity treatment. Therefore, the 50.69 calculations are not impacted.
Detailed Fire Modeling	The application of fire modeling technology is used in the FPRA to translate a fire initiating event into a set of consequences (fire induced failures). The performance of the analysis requires a number of key input parameters. These input parameters include the heat release rate (HRR) for the fire, the growth rate, the damage threshold for the targets, and response of plant staff (detection, fire control, fire suppression). The fire modeling methodology itself is largely empirical in some respects and consequently is another source of uncertainty. For a given set of input parameters, the fire modeling results (temperatures as a function of distance from the fire) are characterized as having some distribution (aleatory uncertainty).	Consensus modeling approach is used for the Detailed Fire Modeling. It is concluded that the methodology for the Detailed Fire Modeling task does not introduce any epistemic uncertainties that would require sensitivity treatment. Therefore, the 50.69 calculations are not impacted.

Fire PRA Sources of Model Uncertainty		
Description	**Sources of Uncertainty**	**Disposition**
	The epistemic uncertainty arises from the selection of the input parameters (specifically the HRR and growth rate) and how the parameters are related to the fire initiating event. While industry guidance is available, that guidance is derived from laboratory tests and may not necessarily be representative of randomly occurring events. The fire modeling results using these input parameters are used to identify a zone of influence (ZOI) for the fire and cables/equipment within that ZOI are assumed to be damaged. In general, the guidance provided for the treatment of fires is conservative and the application of that guidance retains that conservatism. The resulting risk estimates are also conservative.	

Fire PRA Sources of Model Uncertainty		
Description	**Sources of Uncertainty**	**Disposition**
Post-Fire Human Reliability Analysis	The human error probabilities used in the FPRA were adjusted to consider the additional challenges that may be present given a fire. The human error probabilities were obtained using the EPRI HRAC and included the consideration of degradation or loss of necessary cues due to fire. Given the methodology used, the impact of any remaining uncertainties is expected to be small.	The human error probabilities were obtained using the EPRI HRAC and included the consideration of degradation or loss of necessary cues due to fire. The impact of any remaining uncertainties is expected to be small. Further, as directed by NEI 00-04, fire model 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. These results are capable of driving a component and respective functions HSS and therefore the uncertainty of the PRA modeled HEPs are accounted for in the 50.69 application.
Seismic-Fire Interactions Assessment	Since this is a qualitative evaluation, there is no quantitative impact with respect to the uncertainty of this task.	The qualitative assessment of seismic induced fires should not be a source of model uncertainty as it is not expected to provide changes to the quantified FPRA model. Based on the discussion of sources of uncertainty and the discussion above, it is concluded that the methodology for the Seismic-Fire Interactions Assessment task does not introduce any epistemic uncertainties that would require sensitivity treatment. Therefore, the 50.69 calculations are not impacted.

Fire PRA Sources of Model Uncertainty		
Description	**Sources of Uncertainty**	**Disposition**
Fire Risk Quantification	As the culmination of other tasks, most of the uncertainty associated with quantification has already been addressed. The other source of uncertainty is the selection of the truncation limit. However, the selected truncation was confirmed to be consistent with the requirements of the PRA Standard.	The selected truncation was confirmed to be consistent with the requirements of the PRA Standard. Based on the discussion of sources of uncertainty and the discussion above, it is concluded that the methodology for the Fire Risk Quantification task does not introduce any epistemic uncertainties that would require sensitivity treatment. Therefore, the 50.69 calculations are not impacted
Uncertainty and Sensitivity Analyses	This task does not introduce any new uncertainties. This task is intended to address how the fire risk assessment could be impacted by the various sources of uncertainty.	This task does not introduce any new uncertainties. This task is intended to address how the fire risk assessment could be impacted by the various sources of uncertainty. Based on the discussion of sources of uncertainty and the discussion above, it is concluded that the methodology for the Uncertainty and Sensitivity Analyses task does not introduce any epistemic uncertainties that would require sensitivity treatment. Therefore, the 50.69 calculations are not impacted.

Fire PRA Sources of Model Uncertainty		
Description	**Sources of Uncertainty**	**Disposition**
FPRA Documentation	This task does not introduce any new uncertainties to the fire risk.	This task does not introduce any new uncertainties to the fire risk as it outlines documentation requirements. Based on the discussion of sources of uncertainty and the discussion above, it is concluded that the methodology for the FPRA documentation task does not introduce any epistemic uncertainties that would require sensitivity treatment. Therefore, the 50.69 calculations are not impacted.