10 CFR 50.69
10 CFR 50.90
L-2017-156
December 22, 2017

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

Turkey Point Units 3 and 4
Docket Nos. 50-250 and 50-251

Subject: License Amendment Request 254, 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, Florida Power & Light Company (FPL) is requesting amendments to the licenses for Turkey Point Units 3 and 4.

The proposed amendment would modify the Turkey Point 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 Turkey Point Units 3 and 4 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 Reactors 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.

Though routine maintenance updates have been applied, the NRC has previously reviewed the technical adequacy of the Turkey Point Probabilistic Risk Assessment (PRA) model identified in this application for:

- License Amendment Request (LAR) 216, "Transition to 10 CFR 50.48(c) - NFPA 805 Performance-Based Standard for Fire Protection for Light Water Reactor Electric Generating Plants (2001 Edition)," which was approved on May 28, 2015, ADAMS Accession No. ML15061A237

- LAR-229, "Application for Technical Specification Change Regarding Risk-Informed Justifications for the Relocation of Specific Surveillance Frequency Requirements to a Licensee Controlled Program," which was approved on July 16, 2015, ADAMS Accession No. ML15166A320

FPL requests that the NRC utilize the review of the PRA technical adequacy for these applications when performing the review for this application.

FPL requests approval of the proposed license amendment by December 31, 2018 with the amendment being implemented within 90 days.

In accordance with 10 CFR 50.91, a copy of this application, with attachments, is being provided to the designated Florida Official.

This letter contains no regulatory commitments.

If you should have any questions regarding this submittal, please contact Mitch Guth, Licensing Manager, at (305) 246-6698.

I declare under penalty of perjury that the foregoing is true and correct.

Executed on December 22, 2017.

Sincerely,

Thomas Summers
Regional Vice President - Southern Region

Enclosure: Evaluation of the Proposed Change

cc: USNRC Regional Administrator, Region II
USNRC Project Manager, Turkey Point Nuclear Plant
USNRC Senior Resident Inspector, Turkey Point Nuclear Plant
Ms. Cindy Becker, Florida Department of Health

Enclosure
Evaluation of the Proposed Change

TABLE OF CONTENTS

LIST OF ATTACHMENTS

1 SUMMARY DESCRIPTION

The proposed amendment modifies the licensing basis to allow for the implementation of the provisions of Title 10 of the Code of Federal Regulations (10 CFR), Part 50.69, "Risk-Informed Categorization and Treatment of Structures, Systems, and Components for Nuclear Power Reactors." The 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. The structures, systems, and components (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 Florida Power & Light Company (FPL) to improve focus on equipment that has safety significance resulting in improved plant safety.

2.3 DESCRIPTION OF THE PROPOSED CHANGE

FPL proposes the addition of the following condition to the renewed operating licenses of Turkey Point (PTN) Units 3 and 4 to document the NRC's approval of the use 10 CFR 50.69.

> FPL 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 to a seismic probabilistic risk assessment approach).

Prior to implementation, FPL will complete the modifications associated with the NFPA 805 Fire PRA documented in Attachment S of FPL letter 2014-003, "Response to Request for Additional Information Regarding License Amendment Request No. 216 - Transition to 10 CFR 50.48(c) - NFPA 805 Performance-Based Standard for Fire Protection for Light Water Reactor Electric Generating Plants (2001 Edition)," dated November 5, 2014.

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.

Though routine maintenance updates have been applied, the NRC has previously reviewed the technical adequacy of the PTN Probabilistic Risk Assessment (PRA) model identified in this application for:

- License Amendment Request (LAR) 216, "Transition to 10 CFR 50.48(c) - NFPA 805 Performance-Based Standard for Fire Protection for Light Water Reactor Electric Generating Plants (2001 Edition)," which was approved on May 28, 2015 (Reference 2).
- LAR-229, "Application for Technical Specification Change Regarding Risk-Informed Justifications for the Relocation of Specific Surveillance Frequency Requirements to a Licensee Controlled Program," which was approved on July 16, 2015 (Reference 3).

FPL requests that the NRC utilize the review of the PRA technical adequacy for these applications when performing the review for this application.

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

3.1.1 Overall Categorization Process

FPL will implement the risk categorization process in accordance with the 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 4). 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 (mechanical and electrical) 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 the 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 FPL 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.
- 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.
- FPL 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 IDP must intervene to assign any of these HSS Function components to low safety significant (LSS).

- With regard to the criterion that considers whether the active function is called out or relied upon in the plant Emergency/Abnormal Operating Procedures, FPL will not take credit for alternate means unless the alternate means are proceduralized and included in Licensed Operator training.

The following is the exception taken to the NEI 00-04 categorization process:

- NEI 00-04, Section 5.3 states that the seismic safety significance process takes one of two forms. Either the use of a plant-specific seismic PRA or a seismic margin analysis (SMA) that reflects the current as-built, as-operated plant is used to identify SSCs that are safety-significant due to seismic risks. However, in accordance with PTN's approved IPEEE and response to GL 88-20 (Reference 5) and FPL's seismic hazard and screening report in response to the Fukushima accident (Reference 6), PTN completed an alternate screening to a SMA. This screening identified that PTN had no significant seismic hazard susceptibilities and vulnerabilities. Section 3.2.3 of this enclosure provides additional information regarding this exception.

The risk analysis being implemented for each hazard is described:

- Internal Event Risks: Internal events PRA model Revision 10, issued July 16, 2012 accepted by NRC for LAR 216, "Transition to 10 CFR 50.48(c) - NFPA 805 Performance-Based Standard for Fire Protection for Light Water Reactor Electric Generating Plants (2001 Edition)" (Reference 2) with routine maintenance and updates applied.

- Internal Flooding risks: Internal Flooding PRA model Revision 1, issued August 21, 2013, accepted by NRC for LAR-229, "Application for Technical Specification Change Regarding Risk-Informed Justifications for the Relocation of Specific Surveillance Frequency Requirements to a Licensee Controlled Program" (Reference 3) with routine maintenance updates applied.
- Fire Risks: Fire PRA model Revision 0, issued July 10, 2014 accepted by NRC for LAR 216, "Transition to 10 CFR 50.48(c) - NFPA 805 Performance-Based Standard for Fire Protection for Light Water Reactor Electric Generating Plants (2001 Edition)," (Reference 2) with routine maintenance updates applied.
- Seismic Risks: Screened based on alternate approach previously accepted by NRC for IPEEE and Fukushima response.
- Other External Risks (e.g., tornados, external floods, etc.): Using the IPEEE screening process as approved by NRC SER dated November 4, 1998 (Reference 5); 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 7), 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 Arkansas Nuclear One (ANO) Risk-Informed Repair/Replacement Activities (RI-RRA) methodology contained in Reference 8 (ML090930246) consistent with the related Safety Evaluation Report (SER) issued by the Office of Nuclear Reactor Regulation.

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 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 dated December 17, 2014 (Reference 9). 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 PTN for 10 CFR 50.69.

The requirements of 10 CFR 50.69 are consistent with the 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 Vogtle SER, "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.

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. The PRA models credited in this request are the same PRA models credited in LAR 216, "Transition to 10 CFR 50.48(c) - NFPA 805 Performance-Based Standard for Fire Protection for Light Water Reactor Electric Generating Plants (2001 Edition)," which was approved on May 28, 2015 (Reference 2), and LAR-229, "Application for Technical Specification Change Regarding Risk-Informed Justifications for the Relocation of Specific Surveillance Frequency Requirements to a Licensee Controlled Program," which was approved on July 16, 2015 (Reference 3), with routine maintenance and updates applied.

3.2.1 Internal Events and Internal Flooding

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

3.2.2 Fire Hazards

The PTN 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 FPL risk management process ensures that the PRA model used in this application reflects the as-built and as-operated plant for each of the PTN units. Attachment 2 of this enclosure identifies the applicable Fire PRA model.

3.2.3 Seismic Hazards

NEI 00-04 requires a seismic risk analysis, either a plant-specific seismic PRA or a SMA that reflects the current as-built, as-operated plant, to identify SSCs that are safety-significant due to seismic risks. However, FPL is proposing that consideration of seismic risk, given the insights below, is not warranted and would not contribute any unique insights in support of the categorization process for PTN. FPL discussed how it intends to address seismic risk during a public meeting with the NRC on October 11, 2017 (Reference 23).

The seismic risk at PTN was evaluated by a site-specific seismic program (documented briefly in Reference 11 with details provided in References 12, 13, and 14) in response to Generic Letter (GL) 87-02, Verification of Seismic Adequacy of Mechanical and Electrical Equipment in Operating

Reactors, Unresolved Safety Issue (USI) A-46, and GL 88-20, Supplement 4 (Reference 10). The site-specific seismic program used plant walk downs rather than high confidence- low probability failure calculations.

In the IPEEE process, PTN was identified as a reduced scope plant per NUREG-1407, with a review level earthquake of 0.15 g peak ground acceleration (PGA), equivalent to the Safe Shutdown Earthquake (SSE). In its review of the IPEEE (Reference 5), the NRC staff concluded "…FPL's IPEEE process is capable of identifying the most likely severe accidents and severe accident vulnerabilities…"

FPL re-evaluated the PTN seismic risk in its response to post-Fukushima Near Term Task Force recommendation 2.1 (Reference 21). The plant-specific ground motion response spectrum (GMRS) developed by EPRI was compared to the site's SSE to find that GMRS has lower value at spectral frequencies from 1 Hz up to 100 Hz, indicating that the plant design basis already bounds the updated seismic hazard; thus, no further analysis was required. PTN Seismic Hazard and Screening Report control point seismic hazard curves were developed and showed seismic hazard for PTN to be low and bounded by the 0.15g PGA (Reference 6). Following a review of the reevaluated seismic hazard for PTN (Reference 22), the NRC staff found that the GMRS, as well as the confirmatory GMRS developed by the NRC staff, are bounded by the SSE for PTN over the frequency range of 1 to 100 Hz. Therefore, a seismic risk evaluation, spent fuel pool evaluation, and a high frequency confirmation were not merited for PTN.

PTN's approach to seismic for PRA has been reviewed and approved by the NRC for prior application. As noted in Reference 3, "the licensee stated that Turkey Point does not have a seismic PRA that Turkey Point is sited in an area of very low seismicity. The licensee also stated that it recently performed additional seismic walkdowns in response to Near-Term Task Force Recommendation 2.3 to identify and address plant degraded, non-conforming, or unanalyzed conditions, with respect to the current seismic licensing basis and that no operability concerns were identified. The licensee also referenced the NRC staff memorandum on Generic Issue 199, which discusses recent updates to estimates of the seismic hazard in the central and eastern United States. This analysis reported a seismic CDF estimate of 5.9E-6 per year for Turkey Point using the "simple average" approach based on the 2008 U.S. Geologic Survey seismic hazard curves and a bounding estimate of 1 E-5 per year. This supports the licensee's conclusion that seismic risk will not be a significant factor for the Turkey Point SFCP." This approach is also provided in the application for Risk-Informed Completion Times in Reference 24.

Based on the low seismic hazard for PTN, FPL proposes that consideration of seismic risk in the 10 CFR 50.69 categorization process is unnecessary. Because of the low seismic hazard, the expected seismic risk would also be very low. Consequently, seismic considerations would not uniquely contribute significant insights to the 10 CFR 50.69 categorization process. Therefore, the consideration of seismic risk will not be utilized for safety categorization under 50.69.

3.2.4 Other External Hazards

Other external hazards were screened from applicability to PTN Units 3 and 4 per a plant-specific evaluation in accordance with GL 88-20 (Reference 10) and updated to use the criteria in ASME PRA Standard RA-Sa-2009 (Reference 15). Attachment 4 provides a summary of the external hazards screening results. Attachment 5 provides a summary of the progressive screening approach for external hazards.

3.2.5 Low Power & Shutdown

The PTN 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 FPL risk management process ensures that the applicable PRA model(s) used in this application continues to reflect the as-built and as-operated plant for each of the PTN 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, FPL 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 and in the prescribed sensitivity studies discussed in Section 5 of NEI 00-04.

In the overall risk sensitivity studies, FPL will utilize a factor of 3 to increase the unavailability or unreliability of LSS components consistent with that approved for Vogtle in Reference 9. Consistent with the NEI 00-04 guidance, FPL 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 all identified PRA models 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.

The detailed process of identifying, characterizing, and qualitative screening of model uncertainties is found in Section 5.3 of NUREG-1855 (Reference 16) and Section 3.1.1 of EPRI TR-1016737 (Reference 17). 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 were reviewed to identify those that would be significant for the evaluation of this application. If the PTN PRA model used a non-conservative treatment, or methods that are 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 risk calculations were considered key for this application.

Key PTN PRA model specific assumptions and sources of uncertainty for this application were identified and dispositioned in Attachment 6. The conclusion of this review is that no additional sensitivity analyses are required to address PTN 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 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 18) 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 January 2002. In April 2011, a focused peer review was performed assessing the human reliability analysis (HRA) and internal flooding analysis portions of the PRA against the 2009 Standard's requirements. Most recently, a focused peer review was performed in October 2013, to assess portions of the PRA model that had received upgrades: common-cause failure analysis, Level 2 analysis, and interfacing system LOCAs.

The Internal Events PRA model was peer reviewed in 2002 by the Westinghouse Owners Group (WOG) prior to the issuance of Regulatory Guide 1.200. As a result, a self-assessment was conducted by FPL of the Internal Events PRA model in accordance with Appendix B of RG 1.200 Revision 2 (Reference 18) to address the PRA technical adequacy requirements not considered in the 2002 peer review. The Internal Events PRA technical adequacy (including the 2002 peer review and self-assessment results) has previously been reviewed by the NRC in previous applications for transition to NFPA-805 (Reference 2) and relocation of surveillance frequency requirements to licensee control (Reference 3). No PRA upgrades as defined by the ASME PRA Standard RA-Sa-2009 (Reference 15) have occurred to the Internal Events PRA model since conduct of the WOG peer review in 2002 that have not had a subsequent focused peer review to Appendix B of RG 1.200 Revision 2.

The Fire PRA model was subject to a self-assessment and a full-scope peer review conducted in February 2010. A subsequent peer review, performed in March 2012, was a focused scope peer review addressing the FSS, HRA, and PRM technical elements of the Fire PRA.

A finding closure review was conducted on the identified PRA models on October 7, 2017. 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 19) as accepted by NRC in the staff memorandum dated May 3, 2017 (ML17079A427) (Reference 20). The results of this review have been documented and are available for NRC audit.

Attachment 3 provides a summary of the remaining findings and open items, including:

- Open items and disposition from the PTN RG 1.200 self-assessments.
- Open findings and disposition of the PTN peer reviews.
- Identification of and basis for any sensitivity analysis needed to address open findings.

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 PTN 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, US Nuclear Regulatory Commission, March 2009.

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

4.2 NO SIGNIFICANT HAZARDS CONSIDERATION ANALYSIS

FPL proposes to modify the licensing basis 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 (SSCs) 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.

FPL has evaluated whether or not a significant hazards consideration is involved with the proposed amendment(s) 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, FPL 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 amendment 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 amendment 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 amendment does 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 amendment meets 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 amendment.

6 REFERENCES

1. NEI 00-04, "10 CFR 50.69 SSC Categorization Guideline," Revision 0, Nuclear Energy Institute, July 2005.
2. Transition to a Risk-Informed, Performance-Based Fire Protection Program in Accordance with Title 10 of the Code of Federal Regulations Section 50.48(c) (TAC Nos. ME8990 and ME8991)," May 28, 2015 (ML15061A237).
3. NRC letter "Turkey Point Nuclear Generating Unit Nos. 3 and 4 - Issuance of Amendments Regarding Risk-Informed Justifications for the Relocation of Specific Surveillance Requirement Frequency Requirements to a Licensee Controlled Program (TAC Nos. MF3931 and MF3932)," July 16, 2015 (ML15166A320).
4. 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.
5. NRC letter "Generic Letter 88-20, Supplement 4, "Individual Plant Examination for External Events for Severe Accident Vulnerabilities" - Turkey Point Nuclear Plant, Units 3 and 4 (TAC Nos. M83687 and MF83688)," November 4, 1998.
6. FPL letter L-2014-085 "Florida Power & Light Company's, Turkey Point Units 3 and 4 Seismic Hazard and Screening Report (CEUS Sites), Response NRC Request for Information Pursuant to 10 CFR 50.54(f) Regarding Recommendation 2.1 of the Near-Term Task Force Review of Insights from the Fukushima Dai-ichi Accident," March 27, 2014 (ML14106A032).
7. NUMARC 91-06, "Guidelines for Industry Actions to Assess Shutdown Management," December, 1991.
8. ANO SER Arkansas Nuclear One, Unit 2 - Approval of 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 (TAC NO. MD5250), April 22, 2009 (ML090930246).
9. 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 (ML14237A034).
10. 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.
11. FPL letter L-94-107 "Verification of Seismic Adequacy of Mechanical and Electrical Equipment in Operating Reactors Unresolved Safety Issue (USI) A-46 Generic Letter (GL) 87-02," May 5, 1994.
12. FPL letter L-91-336 "Individual Plant Examinations of External Events for Severe Accident Vulnerabilities, Generic Letter 88-20, Supplement 4," December 23, 1991.
13. FPL Letter L-92-222, "Individual Plant Examination of External Events (IPEEE) for Severe Accident Vulnerabilities," Generic Letter (GL) No. 88-20, Supplement 4, August 31, 1992.
14. FPL letter L-93-171 "Verification of Seismic Adequacy of Mechanical and Electrical Equipment in Operating Reactors, Unresolved Safety Issue (USI) A-46 Generic Letter (GL) 87-02," September 15, 1993.

15. ASME/ANS RA-Sa-2009, Standard for Level l/Large Early Release Frequency Probabilistic Risk Assessment for Nuclear Power Plant Applications, Addendum A to RA-S-2008, ASME, New York, NY, American Nuclear Society, La Grange Park, Illinois, dated February 2009.

16. NUREG-1855, Guidance on the Treatment of Uncertainties Associated with PRAs in Risk-Informed Decision Making, March 2009.

17. EPRI TR-1016737, Treatment of Parameter and Model Uncertainty for Probabilistic Risk Assessments, December 2008.

18. 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.

19. 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, (ML17086A431).

20. 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 (ML17079A427).

21. NRC Letter, "Request for Information Pursuant to Title 10 of the Code of Federal Regulations 50.54(f) Regarding Recommendations 2.1, 2.3, and 9.3. of the Near-Term Task Force Review of Insights From the Fukushima Dai-Ichi Accident", March 12, 2012 (ML12053A340).

22. NRC letter "Turkey Point Nuclear Generating, Unit Nos. 3 and 4 - Staff Assessment of Information Provided Pursuant to Title10 of the *Code of Federal Regulations*Part 50, Section 50.54(f), Seismic Hazard Reevaluations for Recommendation 2.1 of the Near-Term Task Force Review of Insights from Fukushima Dai-Ichi Accident and Staff Closure of Activities Associated with Recommendation 2.1, "Seismic" (CAC Nos. M F3709 and MF3710)," January 22, 2016 (ML16013A472).

23. "Summary of October 11, 2017, Meeting with Florida Power and Light Company Regarding Planned License Amendment Requests for St. Lucie Plant Units 1 and 2 and Turkey Point Nuclear Generating Unit Nos. 3 and 4 (EPID L-2017-LRM-0020)," November 6, 2017 (ML17291A045).

24. Florida Power & Light Company letter L-2014-369, "License Amendment Request No. 236 Revision to the Technical Specifications to Adopt Risk Informed Completion Times TSTF-505, Revision 1, 'Provide Risk-Informed Extended Completion Times – RITSTF Initiative 4B'," December 23, 2014 (ML15029A297)

Attachment 1: List of Categorization Prerequisites

A. The PRA model to be used for categorization credits the following modifications to achieve an overall CDF and LERF consistent with NRC Regulatory Guide 1.174 risk limits. Use of the categorization process on a plant system will only occur after the modifications are completed.

   Prior to implementation, FPL will complete the modifications associated with the NFPA 805 Fire PRA documented in Attachment S of FPL letter 2014-003, "Response to Request for Additional Information Regarding License Amendment Request No. 216 - Transition to 10 CFR 50.48(c) - NFPA 805 Performance-Based Standard for Fire Protection for Light Water Reactor Electric Generating Plants (2001 Edition)," dated November 5, 2014.

B. Florida Power & Light Company (FPL) 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). 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 lDP for review and approval. The lDP 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 per Section 3.1.1

Attachment 2: Description of PRA Models Used in Categorization

Unit	Model	Baseline CDF	Baseline LERF
3	Internal Events PRA model Revision 10, issued July 16, 2012	7.18E-7	1.87E-8
3	Internal Flooding PRA model Revision 1, issued August 21, 2013	1.62E-7	8.36E-10
3	Fire PRA model Revision 0, issued July 10, 2014	8.66E-5	5.35E-6
3	**Total**	**8.75E-05**[1]	**5.37E-06**[2]
4	Internal Events PRA model Revision 10, issued July 16, 2012	7.13E-7	1.81E-8
4	Internal Flooding PRA model Revision 1, issued August 21, 2013	1.13E-7	4.11E-10
4	Fire PRA model Revision 0, issued July 10, 2014	7.69E-5	4.85E-6
4	**Total**	**7.77E-05**[1]	**4.87E-06**[2]

Notes:

1. Total CDF meets the RG 1.174 acceptance guideline of <1E-4 per year.
2. Total LERF meets the RG 1.174 acceptance guideline of <1E-5 per year.

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

Finding Number	Supporting Requirement(s)	Capability Category (CC)	Description	Disposition for 50.69
Internal Events PRA Model Findings				
DA-2	DA-7	NOT MET	The test and maintenance probabilities used for individual components are based on actual outage time as collected by the plant. The component outage time was clearly collected over the period of time the plant was in Mode 1, 2, 3. The fault trees and event trees use several crossties from AC power, HHSI, and AFW. In the use of these crossties, the opposite unit components have T&M events. The opposite unit may be in Mode 4, 5, 6 at the time of demand and the desired equipment may have lesser Tech Specs than those assumed for power operation. The T&M event probabilities for the opposite unit components must consider unavailability over the total period of demand, not just during power operation. This can be done at the fault logic level (with house events for OOS) or in the data probabilities. Currently, neither is done. The most important case of this is the DG's. The DG T&M unavailability is about 6E-3 (55 hours per year). If the OOS time for major overhaul were considered, the unavailability would be .03 to .05. Consider revising the T & M event probabilities for the opposite unit components to account for unavailability over the total period of demand. As stated above, this can be done at the fault logic level or in the data probabilities.	Documentation updates are needed to close this finding. The documentation updates will not affect the results. No impact on PRA applications.
HR-A2-	HR-A2 HR-B1,	NOT MET	This HR requires identification, through a review of procedures and practices, those calibration activities	Documentation updates are needed to close this finding. The documentation

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

Finding Number	Supporting Requirement(s)	Capability Category (CC)	Description	Disposition for 50.69
01	HR-B2, HR-C3, HR-I2		that if performed incorrectly can have an adverse impact on the automatic initiation of standby safety equipment. The system notebooks contain a detailed listing of testing and maintenance procedures that were identified for each system, but there is no discussion as to which procedures were determined to have the potential to result in equipment being left in a miscalibrated condition, and which were screened from consideration with the basis for screening. A review of the procedures listed in the system notebooks should be performed to identify those that could result in potential miscalibration events, and provide a justification for those that were excluded from further consideration. For miscalibrations that have the potential to impact multiple systems, ensure that they are treated consistently between both systems, and that appropriate HFEs are listed in all impacted system notebooks. Similar traceability needs to be provided for other test and maintenance procedures that have the ability to render a system/equipment unavailable as well.	updates will not affect the results. No impact on PRA applications.
HR-B2-01	HR-B2	NOT MET	This SR does not allow screening of activities that could simultaneously have an impact on multiple trains of a redundant system or diverse system. In the HHSI system notebook, the following valves are assumed not to be under maintenance while either unit is at power: MOV-*-¬864A, B; *-864C; *-845A, B, C, D; MOV-878A, B; MOV-* 856A, B; * 847C; * 882. Because these valves have the potential to impact BOTH Units, they cannot be screened in this	Documentation updates are needed to close this finding. The documentation updates will not affect the results. No impact on PRA applications.

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

Finding Number	Supporting Requirement(s)	Capability Category (CC)	Description	Disposition for 50.69
			manner. Based on this assumption, these valves would only be worked on while both Units are shutdown, which is probably not realistic. Review the actual test and maintenance procedures associated with these valves and determine when they can be subject to testing or maintenance. If they can be subject to testing or maintenance when either of the Units is shutdown, then a T&M needs to be added into the model as well as consideration for a pre-initiator mis-alignment of the valves, and a post-initiator HRA to re-align if necessary.	
HR-C2-01	HR-C2	NOT MET	There is no provided documentation of the plant-specific or applicable generic operating experience for equipment left unavailable for response in accident sequences. Provide documentation of the review of plant-specific or generic operating experience and confirm that no additional failure mode is required.	Documentation updates are needed to close this finding. The documentation updates will not affect the results. No impact on PRA applications.
HR-D1-01	HR-D1	NOT MET	The human failure event probabilities appear to be evaluated with a systematic process that includes an initial screening value and the identification of risk-significant action for which a detailed analysis through ASEP method is used. Although there appear to be some inconsistencies in the values of the HEF, especially for HEF already existing in previous version of the model. For example, action AHFA0N2BK1 is indicated as a pre-existing action (i.e., not highlighted in Table 3, page 22) with an initial value of 1.10E-3. There is no further discussion of this action (i.e., the action is not	Documentation updates are needed to close this finding. The documentation updates will not affect the results. No impact on PRA applications.

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

Finding Number	Supporting Requirement(s)	Capability Category (CC)	Description	Disposition for 50.69
			indicated in Table 4 at page 27 as one of the action requiring further analysis). Still in Table 5 at page 31 the action has a value of 4.0E-5 (consistently with what is in the model). Another example of inconsistency between the documentation, the HRA Calculator file and the CAFTA model is post-initiator action AHFPAFWTHROT).	
LE-D2-01	LE-D2	NOT MET	Electrical penetration assembly failure modes have been found to be important contributors to overall containment fragility at other large dry PWRs, and in at least 2 instances, tend to be the most limiting in terms of ultimate failure pressure. Additionally, early studies at Sandia National Laboratories have considered the potential impact of very high (beyond design basis) temperatures on elastomer seals (this latter issue is more critical for small volume containments such as BWR Mark I). Perform a scoping assessment of the potential impact of electrical penetration thermal mechanical response to severe accidents. Consider using some of the following References: NUREG/CR-4944, CR-5083, CR-5096, CR-5118, and CR-5334.	Documentation updates are needed to close this finding. The documentation updates will not affect the results. No impact on PRA applications.
QU-3	QU-26 QU-19 HR-21	NOT MET	The quantification of a linked fault tree model involves the proper integration of several files which can affect the results. For example: a. The quantification flag file is used to set logic flag events true or false to represent normal system alignment. At PTN, this flag file is also used to set certain maintenance events false. b. The mutually exclusive file is used to remove	Documentation updates are needed to close this finding. The documentation updates will not affect the results. No impact on PRA applications.

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

Finding Number	Supporting Requirement(s)	Capability Category (CC)	Description	Disposition for 50.69
			cutsets from the results file which contain certain combinations of events representing disallowed maintenance or illogical event combinations (i.e., events for failure to open and spurious opening of the same valve in a single cutset). c. The recovery rule file is used to add recovery events to the cutset results based on the appearance of certain combinations of failure events. At PTN, this process is also used to apply human error factors to the quantification results. Since these files control vital processes during quantification, independent review and thorough documentation is needed to ensure that the quantification results do not exclude valid failure sequences. The current mutually exclusive events file (PTN2KMEE.TXT) was changed as a result of the addition of new T&M events for LC/SWGR HVAC AHUs and Sump Level Indicators. The calculation package includes a description of "add double maintenance events for these basic events to mutually exclusive events." However, no justification for making the events mutually exclusive or specifying the combinations that are mutually exclusive is provided. In addition, the review of the mutually exclusive events file indicates that some complimentary combinations related to AFW pump maintenance may not be included. While this would lead to conservative results due to failure to remove invalid cutsets, the addition of inappropriate mutually exclusive combinations would have the opposite result. Similar errors can be introduced through the recovery file through the inappropriate	

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

Finding Number	Supporting Requirement(s)	Capability Category (CC)	Description	Disposition for 50.69
			application of recovery events to sequences which do not represent the conditions assumed in the HRA analysis. Consider developing a documentation package for the flag file, mutually exclusive events file and the recovery rules which provides the basis of each item in the respective files. Cross-disciplinary review of the flag file and mutually exclusive events file by plant personnel may also be considered.	
QU-8	QU-31	NOT MET	The subtier criteria for a grade 3 on this element considers the following to be indicative of a good understanding of the dominant risk contributors: a. The accident sequence results by sequence, sequence types, and total should be reviewed and compared to similar plants to assure reasonableness and to identify any exceptions. b. A detailed description of the Top 10 to 100 accident cutsets (CAFTA or NUPRA) or accident sequences (RISKMAN) should be provided because they are be important in ensuring that the model results are well understood and that modeling assumption impacts are likewise well known. c. The dominant accident sequence groups or functional failure groups should also be discussed. These functional failure groups should be based on a scheme similar to that identified by NEI in NEI 91-04, Appendix B. There is no discussion of results in the calculation packages for updates provided to the review team to indicate that this type of evaluation is done of the quantification results. Also, the calculation packages	Documentation updates are needed to close this finding. The documentation updates will not affect the results. No impact on PRA applications.

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

Finding Number	Supporting Requirement(s)	Capability Category (CC)	Description	Disposition for 50.69
			provide no discussion of how the dominant cutsets or important systems were affected by the changes to the model when compared to the previous revision. Consider expanding the discussion of the quantification results in the calculation packages or developing a PSA Summary Document containing this type of evaluation for each revision.	
Internal Flood PRA Model Findings				
IFQU-A7-01	IFQU-A7 IFQU-A10 IFQU-B2	NOT MET	This SR states: PERFORM internal flood sequence quantification in accordance with the applicable requirements described in paragraph 4.5.8. The internal flooding analysis has been quantified in accordance with internal events quantification requirements; however, supporting documentation should be provided which describes the process. The quantification process should either be documented in the flooding analysis, or if the same process has been used elsewhere, the flooding analysis should point to that process. Additionally a review of the quantification should be documented.	Documentation updates are needed to close this finding. The documentation updates will not affect the results. No impact on PRA applications.
IFSN-A2-01	IFSN-A2	NOT MET	No identification of flood alarms or floor drains has been made in the flood analysis document. PTN should document and identify the presence of flood alarms and floor drains as related to their treatment in the analysis.	Prior to implementation, either this finding will be closed or a sensitivity study case will be performed to determine the impact on the CDF and LERF results for those categorizations that could be adversely affected by this finding.
IFSN-A4-01	IFSN-A4 IFSN-A9	NOT MET	No supporting information has been provided to justify the estimations regarding flood volumes and	Prior to implementation, either this finding will be closed or a sensitivity study case will

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

Finding Number	Supporting Requirement(s)	Capability Category (CC)	Description	Disposition for 50.69
			the subsequent flooding height. PTN should document the calculations performed in determining flood volumes in a given flood area as it relates to equipment in the room (the floor area the equipment takes up), the capacity of the system, the length of time the flood persists, etc.	be performed to determine the impact on the CDF and LERF results for those categorizations that could be adversely affected by this finding.
IFSN-A6-01	IFSN-A6	NOT MET	This SR States: For the SSCs identified in IF-C2c, IDENTIFY the susceptibility of each SSC in a flood area to flood-induced failure mechanisms. INCLUDE failure by submergence and spray in the identification process. EITHER: a) ASSESS qualitatively the impact of flood-induced mechanisms that are not formally addressed (e.g., using the mechanisms listed under Capability Category III of this requirement), by using conservative assumptions; OR b) NOTE that these mechanisms are not included in the scope of the evaluation. No discussion has been provided for the impact due to the additional flood failure mechanisms. Analysis should be performed which includes failure by submergence or spray, and a qualitative assessment of other failure mechanisms needs to be provided (e.g. jet impingement, pipe whip, humidity, condensation, temperature concerns, and any other identified failure modes in the identification process.) Note that the qualitative assessment is a requirement of the NRC Clarification of this SR.	Prior to implementation, either this finding will be closed or a sensitivity study case will be performed to determine the impact on the CDF and LERF results for those categorizations that could be adversely affected by this finding.

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

Finding Number	Supporting Requirement(s)	Capability Category (CC)	Description	Disposition for 50.69
IFSN-A8-01	IFSN-A8	NOT MET	This SR states: IDENTIFY inter-area propagation through the normal flow path from one area to another via drain lines; and areas connected via back flow through drain lines involving failed check valves, pipe and cable penetrations (including cable trays), doors, stairwells, hatchways, and HVAC ducts. INCLUDE potential for structural failure (e.g., of doors or walls) due to flooding loads. Although the obvious propagation pathways (e.g. doors, stairwells, grating) were identified, a good discussion associated with less obvious pathways (e.g. failed backflow check valves, cable penetrations, cable trays, etc.) for individual zones was not found. Documentation of less obvious possible propagation pathways needs to be addressed.	Prior to implementation, either this finding will be closed or a sensitivity study case will be performed to determine the impact on the CDF and LERF results for those categorizations that could be adversely affected by this finding.
IFSO-A1-01	IFSO-A1	NOT MET	Based on a confirmatory walkdown performed the Peer Review Team, the locations/impacts of some pipes containing water may have been overlooked in the analysis. It is recommended that the analyst ensure that spatial information be captured appropriately for spray concerns. Equipment has been identified in walkdown sheets for elevation, but not spatial location. Additionally the analyst should ensure that all potential fluid sources in a given flood area are identified, and all potentially impacted equipment is identified the impact of it failing is evaluated.	Documentation updates are needed to close this finding. The documentation updates will not affect the results. No impact on PRA applications.
IFSO-A4-01	IFSO-A4 IFEV-A7	NOT MET	No human-induced mechanisms have been included in the analysis, and additionally, no process which justifies their exclusion was provided. it is	Prior to implementation, either this finding will be closed or a sensitivity study case will be performed to determine the impact on

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

Finding Number	Supporting Requirement(s)	Capability Category (CC)	Description	Disposition for 50.69
			recommended that specific instances be discussed as it relates specifically to operator induced failures. Additionally, a process or program should be identified which prevents human-induced floods from occur, thereby justifying their exclusion from the analysis.	the CDF and LERF results for those categorizations that could be adversely affected by this finding.
IFSO-A5-01	IFSO-A5	NOT MET	No summary or characterization of flood sources included in the analysis has been provided. It is difficult to tell what the decisions making up the source characterization were. Characterize flood sources in terms of capacity, flow rate, pressure, temperature, etc. Additionally, document the justification for a given flow rate. PTN should also document the process used to identify potential flood sources.	Prior to implementation, either this finding will be closed or a sensitivity study case will be performed to determine the impact on the CDF and LERF results for those categorizations that could be adversely affected by this finding.
Fire PRA Model Findings				
1-3	AS-B1 ES-A1 ES-A3 ES-A4 FQ-A2	NOT MET	The PRA Assumes a reactor trip rather than mapping the components to all of the previously modeled internal events Initiating events. As a result, the equipment that can cause the various initiating events are not mapped to individual initiating events. The internal events PRA model has numerous locations in the model where the specific initiating event results in a model impact. For example, under gate U3QT07; initiating events that can cause a PORV or SRV to lift are ANDed with the failure to reclose the PORV or SRV. In this case, special initiator %ZZIP6U3 is identified as an initiating event that will cause a PORV lift, along with	Prior to implementation, either this finding will be closed or a sensitivity study case will be performed to determine the impact on the CDF and LERF results for those categorizations that could be adversely affected by this finding.

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

Finding Number	Supporting Requirement(s)	Capability Category (CC)	Description	Disposition for 50.69
			%ZZT2U3. Equipment that can cause each are not mapped or modeled in the Fire PRA. As a result of a previous review, the modeling of Feed-and-Bleed was changed to assume a loss of feedwater (low SG level) occurred. The shorter time results in a higher HEP for feed-and-bleed in all scenarios, regardless of whether a loss of FW occurred. However, numerous other modeling impacts can occur, that are not modeled. Under gate I62115, logic for HVAC unit 3S230 failure to start is included when a Loss of offsite power would occur. This logic is applicable only for when a LOOP occurs, and not applicable for non-LOOP events. This type of logic is contained throughout the internal events PRA modeling. Another example is under gate E1104A, where loss of DC power results in lockout relay failures. There are many other examples throughout the PRA. Additionally, the identification of the specific initiating event for quantification was not performed per the requirements of FQ-A2. For quantification, the modeled initiating event is assumed to be a reactor trip in all cases. This treatment does not meet the intent of SR FQ-A2, where the quantified model should encompass the risk contribution from all applicable initiating events. Map all identified internal events initiating events to the specific components that can cause the event, and modify the FPRA to determine the CCDP based on the fire-induced initiating event that results.	
1-10	IGN-A9	NOT MET	Transient Fires are postulated in all fire	Prior to implementation, either this finding

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

Finding Number	Supporting Requirement(s)	Capability Category (CC)	Description	Disposition for 50.69
			compartments, as listed in Appendix B and Table 3-6 of the Ignition Frequency Report. All factors affecting the fire frequency were assessed based upon a slightly modified NUREG/CR-6850 approach. However, the rankings that were provided do not appear to be consistent with the methods in NUREG/CR-6850, result in an underestimate for fire frequencies in some areas, and an over estimate in other areas. One F&O is provided on this SR. In particular: a) Areas were ranked as zero in maintenance, occupancy, or storage even though entrance to the areas is physically possible, b) Areas were ranked as 1, even though activities were not prohibited by plant procedure. In areas where the room is sealed during operation (roof plugs), transients could have been left in the room prior to sealing, so the ranking on this factor should not be zero - per the 6850 guidance. During the walkdown, Compartments 70 and 71 both had permanently stored breaker grounding devices, with poly-covers, and 71 had a temporary transformer for the polar crane (operating). Both should be ranked as 'medium' for storage. Similarly, the cable room had storage of 3 temporary fans, cables and blankets and should be marked as medium for storage. This room also appears to include numerous components that will likely be worked on during power, (ranking moderate for non-hot work), and numerous people were present during our limited walkdown. Compartment 88, an open area in front of the	will be closed or a sensitivity study case will be performed to determine the impact on the CDF and LERF results for those categorizations that could be adversely affected by this finding.

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

Finding Number	Supporting Requirement(s)	Capability Category (CC)	Description	Disposition for 50.69
			switchgear room, had numerous combustibles stored and located, and should probably be marked as medium or high (presently marked as low). Both area 85 and 88 have frequent foot traffic, and should be marked as medium for occupancy. 85 appears as if it should be moderate for storage (no controls). Similarly; no controls appear to be in place for 116. The above are samples of identified issues, based on our limited walkdown. It appears there will be similar issues with other areas in the plant. We looked at other areas adjacent to the areas we were in (compartments 87, 84, etc), and expect similar problems with the present rankings. (This F&O originated from SR IGN-A9) Re-assess the transient fire rankings per the Guidance in NUREG/CR-6850. Confirm the rankings by walkdown of each area, taking into account the actual condition.	
1-17	IGN-A10 QU-E3 UNC-A1 UNC-A2	NOT MET	Table 3-2 includes uncertainty values (EF) for prior and posterior values. However, Error Factors are not propagated to the compartment specific ignition frequencies. The other parameters, such as conditional failure probabilities for circuit failures, do not have uncertainty intervals. The lack of uncertainty intervals would not generate meaningful uncertainty interval of the CDF/LERF results. (This F&O originated from SR IGN-A10) Estimate EFs for significant fire compartments. ESTIMATE the uncertainty interval of the CDF	Prior to implementation, either this finding will be closed or a sensitivity study case will be performed to determine the impact on the CDF and LERF results for those categorizations that could be adversely affected by this finding.

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

Finding Number	Supporting Requirement(s)	Capability Category (CC)	Description	Disposition for 50.69
			results. ESTIMATE the uncertainty intervals associated with parameter uncertainties (DA-D3, HR-D6, HR-G8, IE-C15), taking into account the state-of-knowledge correlation.	
1-18	IGN-A7	NOT MET	During walkdowns, several key areas appeared to have ignition sources not included on the ISDS. For example, in the cable spreading room, 2 transformers were in the compartment (3X033 - 75KVA, 3X130 - 45KVA), both within the screening distance of targets. Also in the compartment is CP-600 spectralink cabinet, an open cabinet, the RCP Vibration Monitoring Cabinet, 4P21 and 4P09 instrument AC panel. Note; we did not do a 100% review of the CS room, so additional cabinets may be missing. See also F&O 1-19. (This F&O originated from SR IGN-A7) Perform a re-verification of the ISDS for significant fire areas in the FPRA. Add missing components to each ISDS, where applicable.	Prior to implementation, either this finding will be closed or a sensitivity study case will be performed to determine the impact on the CDF and LERF results for those categorizations that could be adversely affected by this finding.
1-19	IGN-A7	NOT MET	It appears the Ignition Source Counting did not count Lighting Panels or other similar panels. For example, there were at least 8 lighting panels in the cable spreading room that were not on the ISDS. Additional similar panels are located in most electrical rooms we walked down, such as the switchgear rooms and other electrical rooms. Based on our walkdowns, many of the lighting panels should be included in the ISDS, based on	Documentation updates are needed to close this finding. The documentation updates will not affect the results. No impact on PRA applications.

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

Finding Number	Supporting Requirement(s)	Capability Category (CC)	Description	Disposition for 50.69
			guidance in 6850 and the subsequent FAQ on sealed cabinets. A review of the generic guidance provided for ignition counting did list the screening of small, wall mounted cabinets (sealed). However, the lighting panels do not appear to meet the criteria listed in the procedure (not sealed, numerous switches/breakers), etc. Many of the cabinets are located close to cable trays or other intervening combustibles, so a small fire could result in a larger fire due to spreading. (This F&O originated from SR IGN-A7) Include unsealed lighting panels and similar electrical cabinets in the ISDS as potential ignition sources.	
1-25	FQ-E1 QU-D5	NOT MET	There does not appear to be a review of non-significant cutsets in the PRA documentation. (This F&O originated from SR QU-D5) Perform a review of non-significant cutsets and accident sequences, as discussed in QU-D5 for the FPRA.	Documentation updates are needed to close this finding. The documentation updates will not affect the results. No impact on PRA applications.
1-27	FQ-E1 LE-F1 LE-F2 LE-F3 UNC-A1	NOT MET	Significant fire compartment contributors to LERF are documented in Appendix C of the summary report. However, the contribution from plant damage states is not provided or the contributors from LE-B SRs. Sources of uncertainty, including sensitivity analysis performed, are not evaluated for LERF. (This F&O originated from SR LE-F1) Document the contributors to LERF based on the requirements of LE-F1 of the internal events section of the standard, as required by FE-Q1. Document the Sources of uncertainty, including	Documentation updates are needed to close this finding. The documentation updates will not affect the results. No impact on PRA applications.

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

Finding Number	Supporting Requirement(s)	Capability Category (CC)	Description	Disposition for 50.69
			sensitivity analysis performed for CDF in Appendix D of the Summary Report.	
1-38	FQ-F1 QU-F2 UNC-A2	NOT MET	Results of the Fire PRA did not include the following: (e) the total plant CDF and contributions from the different initiating events and accident classes (i) the uncertainty distribution for the total CDF (j) importance measure results (l) asymmetries in quantitative modeling to provide application users the necessary understanding of the reasons such asymmetries are present in the model (m) the process used to illustrate the computer code(s) used to perform the quantification will yield correct results process. Some of these issues are listed in other F&Os. However, item e (accident classes), l (asymmetries) and m (validation of computer codes) is not covered elsewhere. (This F&O originated from SR QU-F2) Provide required documentation per QU-F2 and FQ-F1.	Documentation updates are needed to close this finding. The documentation updates will not affect the results. No impact on PRA applications.
3-3	PP-B1 PP-B3	NOT MET	A few cases of special separation are credited in the PB&P. Most notable are separation of Fire Compartments 058 and 037 and 004 and 010. The FHA notes in the write-up for fire zone 004: 'There is a partial height concrete wall on the South side of this room with a full height opening to Fire Zone 10'. No justification is provided for this separation, hence it is not clear that the credited separation may be expected to contain the effects of a fire. Accordingly the effect of a fire beyond the identified	Prior to implementation, either this finding will be closed or a sensitivity study case will be performed to determine the impact on the CDF and LERF results for those categorizations that could be adversely affected by this finding.

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

Finding Number	Supporting Requirement(s)	Capability Category (CC)	Description	Disposition for 50.69
			fire compartment boundary may occur. While this effect would be expected to be identified through performance of the multicompartment analysis the level of documentation provided in support of the PB&P does not satisfy the standard requirements. (This F&O originated from SR PP-B)	
3-5	SF-A1	NOT MET	According to the Section 3.13 of the PTN FPRA Summary Report the effect of an earthquake on ignition source scenarios is discussed in the IPEEE and Potential Fire Related Vulnerabilities self assessment. Review of the Potential Fire Related Vulnerabilities self assessment did not reveal an analysis that specifically addresses generation of fire ignition source scenarios which could result from an earthquake, nor does this assessment address the potential risk significance of these scenarios. This assessment does identify fire vulnerabilities in terms of fuels, ignition sources, and oxidizers however these discussions are not specific to seismic events nor do they include evaluation of special ignition scenarios that may arise from an earthquake. (This F&O originated from SR SF-A1)	Documentation updates are needed to close this finding. The documentation updates will not affect the results. No impact on PRA applications.
5-13	FQ-A3	NOT MET	Turkey Point FPRA Summary Report NUREG/CR-6850 Task 16 Report No. 049306006.005 Rev. 1 Tables A-1, A-2, B-1 and B-2 documented the Units 3 & 4 Fire PRA quantification Results for both CDF and LERF for all fire scenarios that were quantified. Scenario 096-A was randomly picked review for both Units 3 & 4. The CDF/LERF results are consistent between the Summary Report and	Prior to implementation, either this finding will be closed or a sensitivity study case will be performed to determine the impact on the CDF and LERF results for those categorizations that could be adversely affected by this finding.

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

Finding Number	Supporting Requirement(s)	Capability Category (CC)	Description	Disposition for 50.69
			ZoneScenarios in database files, Unit 3 CDF "PTNFIRE_W_LERF_MH_ESF.mdb", Unit 3 LERF "PTNFIRE_W_LERF_MH_ESF.mdb", Unit 4 CDF "U4PTNFIRE_W_LERF_MH_ESF.mdb", and Unit 4 LERF "U4PTNFIRE_W_LERF_MH_ESF.mdb". However, reviewing the AlteredEvents table in each database files shows inconsistent basic events impacted between Unit 3 and 4. Unit 3 have no basic event impacted, while Unit 4 have 9 basic events listed. (This F&O originated from SR FQ-A3)	
6-9	FQ-A4 QU-A3	NOT MET	The parametric uncertainty analysis as discussed in QU-E3 (estimate of uncertainty intervals, etc.) is not performed. Also, the "state-of- knowledge" correlation between fire-specific event probabilities (e.g., suppression system unavailabilities, fire ignition frequencies, hot short conditional probabilities, etc.) hasn't yet been applied. (This F&O originated from SR QU-A3)	Documentation updates are needed to close this finding. The documentation updates will not affect the results. No impact on PRA applications.
6-20	CF-A2 UNC-A2	NOT MET	The parametric uncertainty associated with conditional circuit failure probabilities are not evaluated and are not incorporated into the model. (This F&O originated from SR CF-A2) Develop uncertainty intervals for applied hot short probabilities and include them in the model.	Prior to implementation, either this finding will be closed or a sensitivity study case will be performed to determine the impact on the CDF and LERF results for those categorizations that could be adversely affected by this finding.
7-3	PRM-14	NOT MET	The current model uses the LERF model for the PTN revision 9 model (PTN-BJFR-99-010, Rev. 1) and maps appropriate equipment impacts into the	Documentation updates are needed to close this finding. The documentation updates will not affect the results. No

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

Finding Number	Supporting Requirement(s)	Capability Category (CC)	Description	Disposition for 50.69
			system models used to model LERF. No new accident progressions beyond the onset of core damage were identified for the fire PRA. However, there is no documentation that a specific review of the accident progressions leading to LERF was conducted to identify whether new considerations should be addressed in the fire PRA. In addition, effects on PDS mapping due to fire-induced failures may not be appropriately captured. For example, RWST diversion of the RWST to the containment sump is modeled as a failure of HHSI which would normally go to a dry containment PDS. However, the actual PDS should be one for wet containment. While this is a late containment failure concern rather than a concern for LERF, there may be similar fire induced failures that could affect the mapping of LERF accident progressions. (This F&O originated from SR PRM-B14)	impact on PRA applications.
8-3	PRM-B2	NOT MET	Attachment U – Internal Events PRA Quality (DRAFT), document applicability of Internal Events F&Os to internal events PRA, but not to Fire PRA. There was no evidence that the review of F&O disposition status addressed the question of whether the disposition that was taken would adversely affect the development of the fire PRA. This F&O is derived from 2010 Fire PRA peer review F&O 4-4. (This F&O originated from SR PRM-B2)	Prior to implementation, either this finding will be closed or a sensitivity study case will be performed to determine the impact on the CDF and LERF results for those categorizations that could be adversely affected by this finding.
9-6	FSS-D7	NOT MET	The system unavailability records for the plant have not been reviewed in crediting fire detection and suppression systems. This F&O supersedes 2010 FPRA peer review F&O 2-26 (This F&O originated	Documentation updates are needed to close this finding. The documentation updates will not affect the results. No

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

Finding Number	Supporting Requirement(s)	Capability Category (CC)	Description	Disposition for 50.69
			from SR FSS-D7)	impact on PRA applications.
10-1	FSS-C1 FSS-G1	NOT MET	The 2010 peer review identified that "Fire modeling was conducted via generic fire modeling from which Zones-Of-Influence (ZOI) for specific initiator types was generated. The ZOIs were used to define bounding fire characteristics for each fire scenario. Characteristics that are used to bound potentially risk contributing fire events are identified in Attachment B of the Fire Scenario Report, (Report 0493060006.004). Based on the use of a bounding approach this SR is judged to be met at CC I. Significant fire scenarios should be developed with 2-point fire modeling." Since this review, FP&L has stated that "The use of a panel split fraction to differentiate between fires impacting the panel and components with cables terminating at the panel versus panel fires impacting cables outside of the panel provides an equivalent and more useful two point fire model." The Panel Split fraction is developed from a supplemental report (ERIN report, Supplemental Fire PRA Methods, dated February 2010). This document was submitted to the EPRI Fire PRA Methods Review Panel. This review is not complete as of the date of this peer review. Use of the split fraction method is based on industry events rather than site specific fire ignition sources and target configurations and therefore, could result in nonconservative frequency estimates of target damage. (This F&O originated from SR FSS-C1) Perform 2-point fire modeling, when applicable, for	Prior to implementation, either this finding will be closed or a sensitivity study case will be performed to determine the impact on the CDF and LERF results for those categorizations that could be adversely affected by this finding.

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

Finding Number	Supporting Requirement(s)	Capability Category (CC)	Description	Disposition for 50.69
			risk significant fire scenarios.	
10-2	FSS-A1	NOT MET	The 2010 review of PTN Tasks 8 and 11 Report 0493060006.004, identified that 'no hydrogen fires other than turbine/generator have been postulated.'(Previously F&O 5- 16) Since this Finding was identified, FP&L has determined that 'Miscellaneous Hydrogen piping at PTN is limited to hydrogen supply to the VCT tanks. The associated piping is located in the charging pump rooms (Fire Zones 45 and 55). Fires in these fire zones are assumed to impact all components in the fire zone. The associated risk is low given the availability of thermal barrier cooling for RCP seals and HHSI pumps. Allocation of the IGF associated with miscellaneous hydrogen fires to these fire zones would result in an increase in the ignition frequency for these zones by less than a factor of 3. Given the low risk significance of these zones this will have a negligible impact on overall plant risk and the charging pump rooms will remain low risk contribution fire zones. Incorporation of this ignition frequency into the associated documentation will be incorporated in a future revision to the documentation.' Hydrogen fires are also being developed for H2 piping and valves in Compartments 82 and 87 (scenarios 82-P and 87-P). However, since these do not appear yet in the Fire Scenario Report, action is required. This finding is currently being addressed and appears to be resolved once the new H2 fires are included in the model and documentation is updated. (This F&O originated	Prior to implementation, either this finding will be closed or a sensitivity study case will be performed to determine the impact on the CDF and LERF results for those categorizations that could be adversely affected by this finding.

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

Finding Number	Supporting Requirement(s)	Capability Category (CC)	Description	Disposition for 50.69
			from SR FSS-A1) Incorporate the hydrogen fire scenarios being developed into the model, and update documentation as necessary.	
10-4	FSS-C8	NOT MET	One situation was identified for which credit of fire wrap is taken in Compartment 96 for ignition source 3B04, which is a 480V load center. This fire wrap protects PB3319, PB3813, PB7022, and PB7521. The wrap appears as being credited in a HEAF scenario. No justification for crediting this wrap assuming mechanical damage and direct flame impingement from the HEAF is provided. Similar issue for 3B03 also in Compartment 96. Thermo-lag is also seen as credited in some scenarios, which would require justification due to issues with this particular type of cable barrier. (This F&O originated from SR FSS-C8)	Prior to implementation, either this finding will be closed or a sensitivity study case will be performed to determine the impact on the CDF and LERF results for those categorizations that could be adversely affected by this finding.
10-8	FSS-D4 FSS-H4	NOT MET	Ambient conditions are assumed in the Generic Fire Modeling Treatment Report (prepared by Hughes). Ambient temperature is assumed to be 68°F for all calculations. No technical discussion or justification is provided in the Fire Scenario Report to substantiate that this is a reasonable value for the compartments where this was applied. (This F&O originated from SR FSS-D4) Assess areas where elevated ambient temperatures could be experienced and justify the acceptability of the models used. Otherwise, incorporate elevated ambient temperatures into the zone of influence calculations.	Documentation updates are needed to close this finding. The documentation updates will not affect the results. No impact on PRA applications.

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

Finding Number	Supporting Requirement(s)	Capability Category (CC)	Description	Disposition for 50.69
10-11	FSS-C2 FSS-C3 FSS-G1	NOT MET	The 2010 peer review identified that "fire scenario evaluation tools were developed based on the Generic Fire Modeling Treatments. These walkdown/evaluation tools are based on bounding fires that are assumed to cause target damage at a height above the base fire with the fire burning at peak intensity and without burnout times. Because these tools assume a fire burning at peak intensity and without burnout, this SR is considered met at CC I." Since the review, FP&L has stated that "The use of a panel split fraction to differentiate between fires impacting the panel and components with cables terminating at the panel versus panel fires impacting cables outside of the panel provides an equivalent and more useful two point fire model... The application of the two point treatment to individual fire scenarios is carried through to the MCA/HGL evaluation which addresses the impact of each scenario on MCA." The Panel Split fraction is developed from a supplemental report (ERIN report, Supplemental Fire PRA Methods, dated February 2010). This document was submitted to the EPRI Fire PRA Methods Review Panel. This review is not complete as of the date of this peer review. Use of the split fraction method is based on industry events rather than site specific fire ignition sources and target configurations and therefore, could result in nonconservative frequency estimates of target damage. (This F&O originated from SR FSS-C2) Include fire growth and decay for risk significant fire scenarios.	Prior to implementation, either this finding will be closed or a sensitivity study case will be performed to determine the impact on the CDF and LERF results for those categorizations that could be adversely affected by this finding.

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

Finding Number	Supporting Requirement(s)	Capability Category (CC)	Description	Disposition for 50.69
10-14	FSS-A5	NOT MET	Beyond the Generic Fire Modeling Treatments, the Fire PRA did not include additional detailed fire modeling for most fire compartments. Note 4 (under FSS-A5 of the ASME Standard) states that "once a fire scenario has been 'selected,' this implies that the scenario will eventually be evaluated and/or quantified at a level of detail commensurate with the risk significance of the scenario." (This F&O originated from SR FSS-A5) Consider performing additional detailed fire modeling to provide "reasonable assurance that the fire risk contribution of each unscreened physical analysis unit can be characterized."	Prior to implementation, either this finding will be closed or a sensitivity study case will be performed to determine the impact on the CDF and LERF results for those categorizations that could be adversely affected by this finding.
10-15	FSS-C7 FSS-G1 FSS-H7	NOT MET	PTN credits multiple suppression paths for MCA/HGL evaluation. However, the dependencies have not been evaluated and modeled. For example, fixed suppression and fire brigade response may both rely on a single detection system. (This F&O originated from SR FSS-C7) When multiple suppression paths are credited, perform a review and address any dependencies between suppression and detection systems credited in the MCA/HGL calculation.	Documentation updates are needed to close this finding. The documentation updates will not affect the results. No impact on PRA applications.
10-18	FSS-A1	NOT MET	In at least two cases, transient fire scenarios have not been included in the fire modeling for some compartments (e.g., fire compartments 67 and 68). Per discussion with FP&L the transients may have been excluded based on the dominance of the frequency of fixed scenarios. However, transients should only be excluded when precluded by design.	Prior to implementation, either this finding will be closed or a sensitivity study case will be performed to determine the impact on the CDF and LERF results for those categorizations that could be adversely affected by this finding.

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

Finding Number	Supporting Requirement(s)	Capability Category (CC)	Description	Disposition for 50.69
			Based on the size of these rooms, and the presence of secondary combustibles, transient fires could lead to fire growth and eventually HGL, and therefore should be analyzed. (This F&O originated from SR FSS-A1) Include transient scenarios in all compartments where fire modeling has been employed.	
10-19	FSS-H1	NOT MET	For fire modeling analysis of transient fires, FP&L implements a floor area weighting factor. However, the documentation does not include a graphical representation of the assumed transient locations and boundaries. It is therefore not possible to review (or update) transient fires. Also during review of transient weighting factors it appears to have been double counted in some compartments (e.g., compartment 63). Based on discussion with FP&L this was due to an error in the Excel based spreadsheet tool for transient frequency quantification. This appears to be an isolated case and will be corrected. (This F&O originated from SR FSS-H1) Update documentation to include a graphical representation of transient fire locations and boundaries.	Documentation updates are needed to close this finding. The documentation updates will not affect the results. No impact on PRA applications.
10-20	FSS-A1	NOT MET	The fire modeling analysis of the Turbine Generator (T/G) fires is performed in accordance with Appendix O to NUREG/CR-6850. However, there is no discussion regarding the lack of analysis of the catastrophic T/G fire event, which should consider blade ejection, oil line rupture, and hydrogen explosion. Per discussion with FP&L, the	Documentation updates are needed to close this finding. The documentation updates will not affect the results. No impact on PRA applications.

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

Finding Number	Supporting Requirement(s)	Capability Category (CC)	Description	Disposition for 50.69
			catastrophic fire was discounted since the T/G is located outdoors. While this may not result in hot gas layer formation and structural collapse, a review of the guidance is warranted, and inclusion of this event frequency should as a minimum map to the loss of the T/G and if suppression fails, all equipment within the T/G structure. (This F&O originated from SR FSS-A1) Perform a review of the catastrophic T/G fire in accordance with Appendix O to NUREG/CR-6850, or document the justification for excluding this event at PTN.	
10-21	FSS-C3 FSS-G1 FSS-H2	NOT MET	The supplemental generic Fire Model Treatments: Transient Ignition Source Strength includes an assumption for transient burnout of 12 minutes. This burnout time is based on an assumed fire loading and the 317kW heat release rate, and appears to be optimistic given the uncertainty in transient fire loading. The burnout is then used to develop a zone of influence for thermoplastic targets, based on the thermal response tables in Appendix H to NUREG/CR-6850 for thermoplastic cable at 260°C. Since this resultant vertical zone of influence is used to screen transient scenarios from impacting secondary targets higher than 7.3 feet from the floor, additional justification is needed to demonstrate that a 12 minute fire, and subsequent use of 260°C damage threshold is appropriate for screening purposes. Also noted is that Attachment B to the Fire Scenario Report zone of influence does not reflect the same values recommended by the Generic Fire Model Treatment. As an example, the	Documentation updates are needed to close this finding. The documentation updates will not affect the results. No impact on PRA applications.

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

Finding Number	Supporting Requirement(s)	Capability Category (CC)	Description	Disposition for 50.69
			differentiation between transient Severe and Non-Severe categories is not based on a 317kW fire. This appears to be a documentation issue only. (This F&O originated from SR FSS-C3) Provide additional justification for the applied transient fire analysis as a screening approach. Consider increasing the burnout time and using the NUREG/CR-6850 recommended damage threshold to 205°C to bound uncertainties in fuel loading for transient fires.	

Attachment 4: External Hazards Screening

External Hazard	Screening Result		
	Screened? (Y/N)	Screening Criterion (Note a)	Comment
Aircraft Impact	Y	PS2 PS4	Airport hazard meets 1975 SRP requirements. Additionally, airways hazard bounding analysis per NUREG-1855 is < 1E-6/y.
Avalanche	Y	C3	Not applicable to the site because of climate and topography.
Biological Event	Y	C3, C5	Sudden influxes not applicable to the plant design (closed loop systems for ECWS and CWS). Slowly developing growth can be detected and mitigated by surveillance.
Coastal Erosion	Y	C3	Not applicable to the site because of location.
Drought	Y	C5	Plant design eliminates drought as a concern and event is slowly developing.
External Flooding	Y	PS2	Plant design meets 1975 SRP requirements.
Extreme Wind or Tornado	Y	PS2 PS4	The plant design basis tornado has a frequency < 1E-7/y. The spray pond nozzles (not protected against missiles) have a bounding median risk < 1E-7/y.
Fog	Y	C1	Negligible impact on the plant.
Frost	Y	C3	Not applicable to the site because of location.
Hail	Y	C1 C4	Minimal plant impact, bounded by Missile Analysis

Attachment 4: External Hazards Screening

External Hazard	Screening Result		
	Screened? (Y/N)	Screening Criterion (Note a)	Comment
High Summer Temperature	Y	C1	Plant is designed for this hazard. Associated plant trips have not occurred and are not expected.
High Tide, Lake Level, or River Stage	Y	C3 C4	Not applicable to the site because of location. Impact covered in External Flooding hazard. Plant walkdown addressed drainage and runoff issues.
Hurricane	Y	C4	Impact covered in External Flooding hazard and Extreme Winds Hazard. Plant walkdown addressed drainage, runoff, and missile issues.
Ice Cover	Y	C3	Ice blockage causing flooding is not applicable to the site because of location (no nearby rivers and climate conditions).
Industrial or Military Facility Accident	Y	PS2 C4	Explosive hazard impacts and control room habitability impacts meet the 1975 SRP requirements (RGs 1.91 and 1.78). Homestead Air Force Base impact is addressed by Aircraft Impact.
Internal Flooding	N	None	PRAs addressing internal flooding have indicated this hazard typically results in CDFs ≥ 1E-6/y. Also, the ASME/ANS PRA Standard requires a detailed PRA for this hazard which is addressed in the PTN Internal Flooding PRA.
Internal Fire	N	None	PRAs addressing internal fire have indicated this hazard typically results in CDFs ≥ 1E-6/y. Also, the ASME/ANS PRA Standard requires a detailed PRA for this hazard which is addressed in the PTN NFPA 805 PRA.

Attachment 4: External Hazards Screening

External Hazard	Screening Result		
	Screened? (Y/N)	Screening Criterion (Note a)	Comment
Landslide	Y	C3	Not applicable to the site because of topography.
Lightning	Y	C4 C1	Lightning strikes causing loss of offsite power or turbine trip are contributors to the initiating event frequencies for these events. However, other causes are also included. The impacts are no greater than already modeled in the internal events PRA. PTN designed for lightning.
Low Lake Level or River Stage	Y	C3	Not applicable to the site because of location.
Low Winter Temperature	Y	C1 C5	Extended freezing temperatures are rare, the plant is designed for such events, and their impacts are slow to develop.
Meteorite or Satellite Impact	Y	PS4	Likelihood of a large meteorite, large enough to cause significant plant damage, is very low.
Pipeline Accident	Y	PS4	Explosion of gas pipeline at fossil unit results in a total CDF of less than 1E-7/year.
Release of Chemicals in Onsite Storage	Y	PS2	Plant storage of chemicals meets 1975 SRP requirements.
River Diversion	Y	C3	Not applicable to the site because of location.
Sand or Dust Storm	Y	C1 C5	The plant is designed for such events. Slow developing event.

Attachment 4: External Hazards Screening

External Hazard	Screening Result		
	Screened? (Y/N)	Screening Criterion (Note a)	Comment
Seiche	Y	C3	Not applicable to the site because of location.
Seismic Activity	N	None	NRC approved alternate method to Seismic margins analysis (SMA) was performed for the Individual Plant Evaluation-External Events (IPEEE) and re evaluated under response to Fukushima
Snow	Y	C3	Not applicable to the site because of location.
Soil Shrink-Swell Consolidation	Y	C3	Not applicable to the site because of location. SR structures are founded on bedrock and/or engineered fill.
Storm Surge	Y	PS4	Low probability of occurrence based on IPEEE and post Hurricane Andrew modifications.
Toxic Gas	Y	C4	Toxic gas covered under release of chemicals in onsite storage, industrial or military facility accident, and transportation accident.
Transportation Accident	Y	C1 C3 C4	Bounding analyses used for offsite rail shipment of chlorine gas and onsite truck shipment of ammonium hydroxide. Marine accident not applicable to the site because of location. Aviation and pipeline accidents covered under those specific categories.
Tsunami	Y	C1	Storm surge is bounding

Attachment 4: External Hazards Screening

External Hazard	Screening Result		
	Screened? (Y/N)	Screening Criterion (Note a)	Comment
Turbine-Generated Missiles	Y	PS4	Low probability of turbine wheel failure and low probability of missile impacting safety-related equipment; minimal impact
Volcanic Activity	Y	C3	Hazard is not applicable to Turkey Point Units 3 and 4.
Waves	Y	C1	Storm surge is bounding
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	
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	

Attachment 5: Progressive Screening Approach for Addressing External Hazards

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

Assumption/ Uncertainty	Discussion	Disposition
Ignition source counting is an area with inherent uncertainty; however, the results are not particularly sensitive to changes in ignition source counts. The primary source of uncertainty for this task is associated with the frequency values from NUREG/CR-6850 supplement 1 which result in uncertainty due to variability among plants along with some significant conservatism in defining the frequencies, and their associated heat release rates, based on limited fire events and fire test data.	The conservatism in the ignition frequency data, which is also linked to conservatism in nonsuppression probability data specified in NUREG/CR 6850 appears to introduce a significant conservatism.	This uncertainty has the potential to push more components into the HSS categorization than LSS. Future model enhancement incorporating NUREG 2169 will reduce this uncertainty, but not eliminate it, due to larger data set developing frequencies.
The approach taken for this task included: 1) the use of generic fire modeling treatments in lieu of conservative scoping analysis techniques and 2) limited detailed fire modeling was performed to refine the scenarios developed using the generic fire modeling solutions. The primary conservatism introduced by this task is associated with the heat release rates specified in NUREG/CR 6850.	The employment of generic fire modeling solutions did not introduce any significant conservatism. Detailed fire modeling was only applied where the reduction in conservatism was likely to have a measurable impact. No scenarios were identified which would benefit from detailed fire modeling given the detailed nature of the generic treatments used and the application of multi-point treatments based on split fractions for fires impacting only the ignition source versus fires impacting external targets. The NUREG/CR 6850 heat release rates introduce significant conservatism given the limited fire test data available to define the heat release rates and the associated fire development timeline.	This uncertainty has the potential to push more components into the HSS categorization than LSS. Future model enhancement incorporating NUREG 2178 will reduce this uncertainty, but not eliminate it, due to larger data set developing frequencies.