FPL

DEC 1 9 2017

10 CFR 50.90
10 CFR 50.69
L-2017-155

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

St. Lucie Units 1 and 2
Docket Nos. 50-335 and 50-389

Subject: Application to Adopt 10 CFR 50.69, "Risk-Informed Categorization and Treatment of Structures, Systems, and Components for Nuclear Power Reactors"

In accordance with the provisions of 10 CFR 50.69 and 10 CFR 50.90, Florida Power & Light Company (FPL) is requesting amendments to the licenses for St. Lucie (PSL) Unit 1 and Unit 2.

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

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

- License amendment request "Transition to 10 CFR 50.48(c) - NFPA 805 Performance-Based Standard for Fire Protection for Light Water Reactor Generating Plants (2001 Edition)", which was approved on March 31, 2016, ADAMS Accession No. ML15344A346

- License amendment request "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 June 22, 2015, ADAMS Accession No. ML15127A066

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 Mike Snyder, Licensing Manager, at 772-467-7036.

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

Executed on December 19, 2017.

Sincerely,

Daniel DeBoer

Daniel DeBoer
Site Director
Florida Power & Light Company

Enclosure: Evaluation of the Proposed Change

cc: USNRC Regional Administrator, Region II
USNRC Project Manager, St. Lucie Nuclear Plant, Units 1 and 2
USNRC Senior Resident Inspector, St. Lucie Nuclear Plant, Units 1 and 2
Ms. Cindy Becker, Florida Department of Health

Enclosure
Evaluation of the Proposed Change

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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 conditions to the renewed operating licenses of St. Lucie (PSL) Units 1 and 2 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 from a seismic margins approach to a seismic probabilistic risk assessment approach).

Prior to implementation, FPL will implement the modifications to its facility as described in Table S-1 , "Plant Modifications Committed," Attachment S, of Florida Power & Light letter L-2017-058, dated May 2, 2017, to complete the transition to full compliance with 10 CFR 50.48(c).

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 10 CFR 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 PSL Probabilistic Risk Assessment (PRA) model identified in this application for:

- License amendment request “Transition to 10 CFR 50.48(c) - NFPA 805 Performance-Based Standard for Fire Protection for Light Water Reactor Generating Plants (2001 Edition)”, which was approved on March 31, 2016 (Reference 2).
- License amendment request “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 June 22, 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 PSL'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), PSL completed an alternate screening to a SMA. This screening identified that PSL 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 version 17R0 May 2017, accepted by the NRC as version 12R0 May 2012 with routine maintenance and updates applied, and internal flood PRA model 12R0_EPU November 2013 for the following applications:
  - License amendment request "Transition to 10 CFR 50.48(c) - NFPA 805 Performance-Based Standard for Fire Protection for Light Water Reactor Generating Plants (2001 Edition)", which was approved on March 31, 2016 (Reference 2).
  - License amendment request "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 June 22, 2015 (Reference 3).
- Fire Risks: Fire PRA model version 12R0A_EPU-FIRE October 2014 used in license amendment request "Transition to 10 CFR 50.48(c) - NFPA 805 Performance-Based Standard for Fire Protection for Light Water Reactor Generating Plants (2001 Edition)", which was approved on March 31, 2016 (Reference 2) with routine maintenance and updated 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 the NRC SER dated January 25, 1999 (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 PSL 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 license amendment request "Transition to 10 CFR 50.48(c) - NFPA 805 Performance-Based Standard for Fire Protection for Light Water Reactor Generating Plants (2001 Edition)", which was approved on March 31, 2016 (Reference 2), and license amendment request "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 June 22, 2015 (Reference 3), with routine maintenance and updates applied.

3.2.1 Internal Events and Internal Flooding

The PSL 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 PSL units. Attachment 2 of this enclosure identifies the applicable internal events and internal flooding PRA models.

3.2.2 Fire Hazards

The PSL 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 PSL 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 PSL. 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 PSL was evaluated by a site-specific seismic program (documented briefly in Reference 10 with details provided in References 11, 12, and 13) 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 of a Low Probability of Failure (HCLPF) calculations.

In the IPEEE process, PSL was identified as a reduced scope plant per NUREG-1407, with a review level earthquake of 0.10 g peak ground acceleration (PGA), equivalent to the Safe Shutdown Earthquake (SSE).

FPL re-evaluated the PSL 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. PSL Seismic Hazard and Screening Report control point seismic hazard curves were developed and showed seismic hazard for PSL to be low and bounded by the 0.10g PGA (Reference 6). Following a review of the reevaluated seismic hazard for PSL (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 PSL 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 PSL.

PSL's approach to seismic for PRA has been reviewed and approved by the NRC for prior application "Application for Technical Specification Change Regarding Risk-Informed Justifications for the Relocation of Specific Surveillance Frequency Requirements to a Licensee Controlled Program". As noted in Reference 3, "the licensee stated that it does not have a seismic PRA or seismic margin analysis and that St. Lucie 1 and 2 are sited in an area of very low seismicity. The licensee also stated, "Staff at St. Lucie 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, "No operability concerns were identified." In its letter dated December 11, 2014, the licensee responded to APLA RAI 3 and explained that there is no impact to the St. Lucie 1 and 2 SFCP from revised seismic risk values based on the "plant level high confidence of a Low Probability of Failure (HCLPF) using plant-specific Ground Motion Response Spectra (GMRS) that was recently developed by EPRI." This supports the licensee's conclusion that seismic risk will not be a

significant factor for the St. Lucie 1 and 2 SFCP." This approach is also provided in the application for Risk-Informed Completion Times in Reference 24.

Based on the low seismic hazard for PSL, 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 PSL Units 1 and 2 per a plant-specific evaluation in accordance with GL 88-20 (Reference 14) and updated to use the criteria in ASME PRA Standard RA-Sa-2009. 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 PSL 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 PSL 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 15) and Section 3.1.1 of EPRI TR-1016737 (Reference 16). 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 PSL 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 PSL 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 PSL 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 17) consistent with NRC RIS 2007-06.

The internal events PRA model was subject to:

- a full-scope peer review in July 2002.
- a Self-Assessment to RG 1.200 October 2007
- a focused-scope peer review (LERF) July 2009
- a focused-scope peer review (CCF) August 2009

- a Self-Assessment in November 2009
- a focused -scope peer review (DA, HR) in April 2011
- a focused -scope peer review (ISLOCA) in 2013
- a Self-Assessment to RG 1.200 Rev 2 March 2014

The Internal Events PRA model was peer reviewed in 2002 by the Combustion Engineering Owners Group (CEOG) 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 17) 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 20) have occurred to the Internal Events PRA model since conduct of the CEOG peer review in 2002 that have not had a subsequent focused peer review to Appendix B of RG 1.200 Revision 2.

The internal floods PRA model was subject to a focused scope peer review in April 2011.

The Fire PRA model was subject to:

- a full-scope peer review of the PSL Fire PRA model June 2011.
- a focused-scope peer review (FSS) May 2013.
- a focused-scope peer review (FQ) June 2013.

A finding closure review was conducted on the identified PRA models on September 14, 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 18) as accepted by NRC in the staff memorandum dated May 3, 2017 (ML17079A427) (Reference 19). 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 PSL RG 1.200 self-assessment
- Open findings and disposition of the PSL 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 PSL 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 10 CFR 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. NRC letter "St. Lucie Plant, Unit Nos. 1 and 2 - Issuance of Amendments Regarding 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) (CAC Nos. MF1371 and MF 1374)," March 31, 2016 (ML15344A346).
3. NRC letter "St. Lucie Plant, Unit Nos. 1 and 2 - Issuance of Amendments Regarding Risk-Informed Justifications for the Relocation of Specific Surveillance Frequency Requirements to a Licensee-Controlled Program (TAC Nos. MF3495 and MF 3496)," June 22, 2015 (ML15127A066).
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" - St. Lucie Plant, Units Nos. 1 and 2 (TAC NOs. M83678 and M83679)," January 25, 1999.
6. FPL letter L-2014-089 "Florida Power & Light (FPL) 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-Tern Task Force Review of Insights from the Fukushima Dai-ichi Accident," March 31, 2014 (ML14099A106)

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) (ML090930246), April 22, 2009.
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. 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.
11. FPL letter L-91-336 "Individual Plant Examinations of External Events for Severe Accident Vulnerabilities, Generic Letter 88-20, Supplement 4," December 23, 1991.
12. 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.
13. 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.
14. 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.
15. NUREG-1855, Guidance on the Treatment of Uncertainties Associated with PRAs in Risk-Informed Decision Making, March 2009.
16. EPRI TR-1016737, Treatment of Parameter and Model Uncertainty for Probabilistic Risk Assessments, December 2008.
17. 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.
18. 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).
19. 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).
20. 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.
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 "St. Lucie Plant, Units 1 and 2 - Staff Assessment and Closure of Information Provided Pursuant to Title 10 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 the Fukushima Dai-Ichi Accident (CAC No.s M F3940 and MF3941)," January 7, 2016 (ML15352A053).

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-2014-242, "Application to Adopt TSTF-505, Revision 1, Provide Risk-Informed Extended Completion Times - RITSTF Initiative 4B," December 5, 2014 (ML14353A016)

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 implement the modifications to its facility as described in Table S-1 , "Plant Modifications Committed," Attachment S, of Florida Power & Light letter L-2017-058, dated May 2, 2017, to complete the transition to full compliance with 10 CFR 50.48(c).

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
1	Internal Events PRA Model U117R0 March 2017	2.64E-06	2.46E-07
1	Internal Flood PRA Model U112R0_EPU November 2013	1.57E-07	5.65E-07
1	Internal Fire PRA Model U112R0A_EPU-Fire October2014	8.39E-06	3.95E-06
1	**Total**	**1.12E-05**[1]	**4.76E-06**[2]
2	Internal Events PRA Model U217R0 March 2017	1.98E-06	9.58E-08
2	Internal Flood PRA Model U212R0_EPU November 2013	1.23E-06	1.50E-07
2	Internal Fire PRA Model U212R0A_EPU-Fire October 2014	3.72E-06	1.75E-06
2	**Total**	**6.9E-06**[1]	**2.00E-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				
AS-04	AS-A5	Not Met	RWT rupture is assumed to fail shutdown cooling. This seems overly conservative. Without make-up, the level in the RCS would drop, but there is more than enough fluid in the Boric Acid Tanks and the VCT to restore this level. The level does not need to be fully restored to allow shutdown cooling. The level need only be above the hot leg. Estimated Level Drop 2250 psia at 600 F (0.0217 ft^3/lbm) to 100 psia at 300F (0.01766 ft^3/lbm). Given RCS liquid volume of 10,400 ft^3, this means approximately 18,500 gallons are required to restore the PRZ level. Each Boric Acid Tank contains 9700 gallons the VCT contains 4000 gallons. Fully PRZ level is not required full shutdown cooling when core damage is the alternative.	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.
AS-06	AS-A3	Not Met	Consider adding low pressure feed (using Condensate pumps) to the model for accident sequences involving loss of all MFW/AFW. Using condensate pumps to feed the SG's is in both EOP 6 'Total Loss of Feed' and EOP-15 'Functional Recovery Procedure'. Operations is directed to use low pressure feed in 1-EOP-06 (Step 8.B.3.1). Crediting low-pressure feed will eliminate those core damage sequences where the MFW pumps are lost, but the condensate pumps are available. If the TBVs are not available, then the hot well make-up control system (or an operation action) must be modeled to incorporate this alternative. Adding LPF could reduce dependency on Once Through Cooling for a number of accident sequences. (See F&O AS-03 also)	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.
AS-08	AS-A5	Not Met	Check Valves 09294 and 09252 are common for both AFW, MFW, and Low Pressure Feed. These CKVs currently appear only in the AFW system. The may be some events (e.g. LOL) where the turbine trips and steam generator pressure rises enough to cause the closure of these check valves. Under these scenarios, the failure of both of these checks would fail all secondary side heat removal. Currently, these CKVs are modeled under FMM1SGCVLV. This	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
			event has a failure probability far lower than several three element CKV groups in the AFW system. There does not appear to be a basis for this difference. The failure likelihoods (independent and common cause) of the check valves in the AFW system should be consistent or the basis for the difference is documented. Further, as the random failure of these CKVs could cause a LOFW trip and eliminate all secondary side feed to a single S/G, this is worthy of consideration as an initiating event.	
AS-12	AS-A5	Not Met	Currently, shutdown cooling is credited as a long-term cooling method to eliminate the re-circulation requirement on certain ranges of LOCA breaks. A certain amount of water must be above the bottom of the hot leg to avoid drawing vapor into the shutdown cooling system. Some calculation must be done to ensure that the RCS will be above this critical point. This calculation could be quite simple: determine the RCS water level at the point of shutdown cooling entry conditions, determine the leakage rate at the point, verify the RCS level will be adequate for the remaining part of the 24 hr mission without re-circulation or RCS make-up. If this is not true, then addition make-up must be modeled through the emergency sump or CVCS.	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.
AS-13	AS-A2	Not Met	The PORVs are only assumed to lift given total loss of secondary side heat removal or a loss of load with no anticipatory trip. This appears non-conservative. The only loss of load trips considered are TT and loss of off-site power trips. This is based on an informal calculation that shows the RCS pressure exceeds 2300 psia, but stays below the PORV open set point of 2400 psia. This does not consider variations in the time delay between the turbine trip and the reactor trip nor does it consider variations in the PRZ pressure set point. Consideration of these variables may lead the analyst to conclude that	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
			the likelihood of a PORV lift during this condition is much larger than analyzed. Further, the portion of the tree (under Gate U1QT99) that models the circuitry associated with the anticipatory trip only contains a single basic event. No other support system dependencies appear. For example, does the status of pressurizer spray affect this calculation? Are there support system failures that could cause a loss of load and disable or degrade the anticipatory trip function?	
DA-C14-01	DA-C14	Not Met	It appears that the treatment of coincident unavailability for inter-systems was considered. However, there was no clear documentation to demonstrate such treatment. Coincident unavailability due to maintenance for different trains of the same system (intra-system) is not allowed by established plant procedures. Therefore, the calculation of coincident unavailabilities for intra-systems as a result of planned and repetitive activities was not calculated. There was no clear documentation on the treatment of coincident unavailability for inter-systems. Discussion with the utility PRA staff indicated that review of the plant operating history was performed to identify potential coincident unavailability for inter-systems. No such unavailabilities were identified. The PRA staff also demonstrated that the PRA model accounts for coincident unavailability for inter-systems by the use of appropriate mutually exclusive logic.	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	Some inconsistencies have been identified between the documentation, the HRA calculator file and the CAFTA model; one example is AHFL1CSTIV, which is indicated as 2.7E-5 in the summary table 3.0 while appears to have the floor value from ASEP in the HRA calculator file (i.e., 1E-5) (See F&O HR-D1-01). A more conservative value has ben entered in the model, with respect of what the HRA calculator provides.	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
HR-I2-01	HR-I2	Not Met	This SR is associated with the documentation of the process used to identify, characterize, and quantify the pre-initiator, post-initiator, and recovery actions considered in the PRA including the inputs, methods and results. Although the overall HRA analysis looks very good, the current documentation for the dependency analysis and treatment of post-initiator HRAs is incomplete. The current documentation only states that all post-initiator HEPs are set to 1.0 and then fed into the HRA calculator to determine the dependency between the HEP events. There is no discussion of how the rest of the process is performed, including how the HRA Recovery File is used to "reset" combination events to the appropriate values based on the dependency analysis, no discussion on why the HEP values in the BE file are set to 0.5, no discussion of how the HRA calculators dependency analysis was validated, etc. Additionally, there is no assurance that all HEP combinations have been identified and evaluated - see F&O HR-G6-01 for more detail.	Documentation updates are needed to close this finding. The documentation updates should not affect the results. No impact on PRA applications.
HR-I3-01	HR-13	Not Met	There is no discussion on model related uncertainties for pre-initiator HRA calculations. For post-initiator HFE, the EF indicated in Table 9 are then not propagated in the CAFTA file. It is therefore not clear how the uncertainty parameters are treated in the model. A complete uncertainty assessment involves both stochastic uncertainties (included in the HRA calculator) and epistemic (model) uncertainties. A discussion on the assumptions made in the analysis and their potential for impact on the HEP calculations is required to meet the SR. The inconsistency between the EF discussed in the post-initiator HRA notebook (table 9 in Section 3.3) and the actual CAFTA file does not allow a correct uncertainty analysis. Table 9 states that generic Error Factors are used, but there are no error factors in the BE file, so it is unclear how the error factors are propagated. Also, the Combination events, and the renamed post-initiator single events are	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
			not included in the BE file so it is unclear how their error factors are included in the analysis - or if they are even considered.	
IE-C6-01	IE-C6	Not Met	A screening approach is utilized for some lines based on low frequency but this is not quantified. The SR indicates a frequency expectation for screening. Define the estimate for the lines screened on low frequency and show that the calculated frequency supports screening.	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.
IE-C9-01	IE-C9	Not Met	The fault tree model used for the ISLOCA paths assumes that the status of all valves is known when the plant is brought online and the corresponding exposure time is the refueling interval. However, based on discussions with knowledgeable staff, there is no positive means to know that more than one isolation valve is actually holding. Use of status lights is not definitive since there is a +/-5% margin between light changing and valve seating. The exposure time should be based on a positive flow test which may not occur on a refueling basis but based on other studies could be as much as the life of the plant.	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.
QU-02	AS-C3	Not Met	A lot of results sections in the quantification report are blank with a “later” in place of the table or results.	Documentation updates are needed to close this finding. The documentation updates will not affect the results. No impact on PRA applications.
QU-04	AS-B5	Not Met	No uncertainty analysis has been performed on the results from Unit 1 or Unit 2 quantification results.	Documentation updates are needed to close this finding. The documentation updates will not affect the results. No impact on PRA applications.
SL-CCF-12	IE-A6	Not Met	The CCF of ICW traveling screen plugging and the CCF of ICW strainers plugging as contributors to the loss of ICW initiator fault tree are missing from the model and no explanation for their absence is provided. Common cause contributors to the loss of ICW are	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

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
			judged to be both credible and potentially risk-significant. This judgment is based on the failure of a intake screen reported in LER 84-09, 1011/84 (Unit 1), the fact that these issues are addressed in the plant Off-Nominal Operating Procedure 064030, and that data is available for both of these failures in the NRC CCF database. Given that common cause is likely to be a dominant contributor to the loss of ICW and that the nominal loss of ICW frequency is judged to be very low (~1E-5/rx-yr), the modeling of the loss of ICW initiating event is judged to not meet SR IE-A6 for any CC level on CCF. Basis for Significance: This F&O was assigned a Significance of A due to its potential risk significance. Additional Discussion: This finding applies to both St. Lucie Unit 1 and 2 PRA models.	LERF results for those categorizations that could be adversely affected by this finding.
SY-12	AS-A10	Not Met	It appears that in general key control systems in the St. Lucie Plant are not modeled. In the fault tree the AFW flow control system is demanded 3 times, but the basis for using 3 demands is unclear. No analysis has been done to determine the number of cycle the AFW system will undergo. Further, the common cause MOV demand failure rate does only considers a single demand. The model does not differentiate between an overfill and underfill. Overfills in general could lead to the failure of the turbine driven AFW pumps. Note: If the MOVs are demanded twice, it is doubtful that the failure likelihood would double. But it is also clear the failure likelihood will increase. Given the importance of the AFW MOVs, any increase to the failure rates can be quite significant.	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.
SY-15	DA-A4	Not Met	The implementation of the Alpha Parameter methodology for common cause analysis has resulted in conditions that appear to be an over estimation of the contribution from common cause and results	Prior to implementation, either this finding will be closed or a sensitivity study case will be performed to

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
			that do not make obvious sense (i.e. cutsets in which the common cause failure of three check valves [three AFW pump discharge check valves] is more likely than the common cause failure of two check valves [two MFW check valves to the steam generators I-V09294 and I-V09252]). The implementation of the methodology includes an assumption in the development of the parameters of staggered testing. This assumption may be non-conservative. The common cause failure of the check valves in the pump recirculation lines was not considered and justification provided for not including them was not included. Some of the issues may be the result of the use of component specific and generic alpha parameter data.	determine the impact on the CDF and LERF results for those categorizations that could be adversely affected by this finding.
Internal Flood PRA Model Findings				
IFEV-A7-01	IFEV-A7	Not Met	The consideration of human-induced floods was not included in the internal flooding evaluation. EPRI report 1013141 that provided generic data for flood initiating event frequencies stated that "Human induced causes of flooding that do not involve piping system pressure boundary failure such as overfilling tanks and inappropriate valve operations that release fluid from the system are not included."	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.
IFPP-B1-01	IFPP-A5	Not Met	The documentation associated with the plant partitioning is scattered between the initial portion of the document and the walkdown report in Attachment B, which in reality is a discussion of the screening of main structures such as major structures. The walkdown report does not include any explanatory picture and mixes the definition of the area and their screening, without spelling out the generic criteria used for the screening of specific structures. This organization of the information is prone to confusion; moreover, since the area identification and the screening are mixed, some overlook have been noticed. For example, the walkdown notes explicitly mention which bldg has been walked down and the DG BLDG is not listed among those, still, the screening of the DG	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
			BLDG is only discussed in the walkdown report and they are all screened out on the basis that there is no service water (DG are air cooled); there is nevertheless no mention of the potential spray effects of Fire Protection system on a single DG (FP lines have been noticed during the walkdown that may have the potential to spray on the DG cabinet). While the screening of the DG BLD may still be possible (FP lines may be dry since there are large FP valves immediately outside of the DG building that may be deluge valve, or the DG AOT may be sufficient to recover from a spray event on the DG cabinet), the presence of a flood source that has the potential for impacting PRA equipment needs to be addressed. The screening out of the Turbine Building is another example of screening process inconsistent with the screening criteria provided in the standard. While it is true that the TB BDLG is open, a rupture in the condenser expansion joints will induce an initiating event and for this reason the area cannot be screened out for flood considerations. The flood scenario generated by a rupture of the condenser expansion joint may be screened for other reasons (e.g., it may be folded into an already existing IE category with identical plant effect but higher IEF), still a discussion of the reasoning and of the screening criteria needs to be provided. Finally, section 4.1.1 points to the walkdown notes but incorrectly indicating Attachment C rather than Attachment B.	
IFQU-A1-02	IFQU-A1	Not Met	Some inconsistencies in the mapping between the flood events and the basic events associated with impacted equipment has been identified. According to table 20, the top cutsets have to do with ATWS induced by a spray event on the reactor trip switchgear. Spray on the reactor trip switchgear would result in loss of power, which would result in the trip itself. Therefore, even though the reactor trip switchgears are actually impacted by the spray event, the flood initiator needs not to be mapped with the basic event associated 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
			the switchgear because their failure is in the direction of the success. Another example of suspect inconsistency in the mapping is observed from the review of the main CDF contributors. A spray from room 1RAB43-59/58 is not expected to impact both trains since the originating room only hosts 1 train of batteries	
IFSO-A4-01	IFSO-A4	Not Met	No evidence was provided to indicate that human-induced mechanisms were considered to determine their impact as potential sources of flooding. The flooding notebook indicated that the EPRI guideline, as documented in report 1019194, was used in performing the flooding analysis. The EPRI Guidance identified the flooding mechanism that would result in a release. No evidence could be found on the treatment of human-induced flooding. It appears that only pipe failures were considered as flooding mechanism.	Documentation updates are needed to close this finding. The documentation updates will not affect the results. No impact on PRA applications.
IFSO-A6-01	IFSO-A6	Not Met	Confirmatory walkdown to assess the accuracy of the information associated with the source identification and scenario definition were not performed. One walkdown was performed before the identification of the flood source began but flood sources have not been confirmed during a dedicated confirmatory walkdown. Some potential inconsistencies between the isometric drawings used for the identification of the flood sources and actual configuration has been observed during the peer review walkdown. For example Appendix A indicates more than 138' of CC piping in the U2 Battery room A (2RAB43-35) but no CC piping has been observed in the room during the walkdown. On the other hand, demin water lines to the emergency eyewash have been observed during the peer review walkdown in the battery room, which are not listed in the Appendix A datasheet. 2RAB43-36 also does not show DW lines although it is expected that eyewash station are also present and they are indeed shown in the architectural drawing). In	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
			2RAB43-36, the batteries are mentioned to be potentially vulnerable to spray from fire protection but no fire protection is listed as potential source in the room. Appendix A shows multiple examples of datasheet being incomplete even for critical rooms such as the ECCS rooms (see for example 2RAB-10-16B) that would challenge the selection of impacted equipment.	
Fire PRA Model Findings				
CS-A3-01	CS-A3	Not Met	4kV power and 125VDC control cables required to support the operation of the Containment Spray Pump were not identified. Fire PRA Plant Response model and other Fire PRA support tasks are adversely affected. Perform a comparison of the components identified on the MSO (multiple spurious operation) list against the Fire PRA components for which new cable selection was performed (i.e., components not previously identified on the Appendix R safe shutdown equipment list). Verify that the cable selection for the common components supports all credited operations. Fire PRA Plant Response model and other Fire PRA support tasks are adversely affected.	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.
CS-B1-01	CS-B1	Not Met	No evaluation was performed to verify that the new components and cables associated with the Fire PRA is bounded by the existing overcurrent coordination analysis. The evaluation was not completed at this time.	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.
ES-D1-01	ES-D1	Not Met	PI-03-003 provides instruction for circuit analysis to include review of interlocks, instrumentation, and support system dependencies. Cable routing database was reviewed and confirmed that interlocks,	Prior to implementation, either this finding will be closed or a sensitivity study case will be performed to

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
			instrumentation, and support system cables were included in equipment effects. However, demonstration of a review of power supplies, etc. was not readily apparent in the Component Selection report. The development of the Fire PRA equipment list inherently considers the entire component and its supporting equipment; however, it is important to document this information to support peer reviews and applications. It is suggested that document the review to show the interlocks, power supplies, etc. are included (or Referenced) in the development of the Component Selection section. The equipment selection report states that SSEL equipment required to place the plant in hot standby, the PRA end state, are included in the analysis while equipment only associated with taking the plant to cold shutdown were excluded from analysis. No information is provided to facilitate the assignment of individual SSEL instrumentation to specific plant states, which complicates review against this SR. Expand Component and Cable Selection tables to allow SSEL components to be associated with specific plant states. Components are linked to fault tree Basic Events, but suggest document all potential fire induced sequences are confirmed to be associated with a reactor trip initiating event in the fault tree. Improve component selection report to address items identified in this F&O.	determine the impact on the CDF and LERF results for those categorizations that could be adversely affected by this finding.
FSS-A1-01	FSS-A1	Not Met	PSL did not postulate hydrogen (H2) fires other than the turbine generator H2 fires. PSL used the basis that their H2 piping contains excess flow check valves. However, this will not prevent H2 fires. It's likely that plants experiencing H2 fires that contributed to the "potentially challenging" fire frequency also had excess flow check valves. Recommend either postulating H2 fires or developing a	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

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
			stronger technical justification for their exclusion. PSL did not appear consider all pump lube oil fire scenarios (e.g., AFW pumps, Charging Pumps, HPSI pumps, LPSI pumps, MFW pumps, etc.). These scenarios often involve significant quantities of oil causing widespread damage in the fire compartment. They can also contribute to multi-compartment fire risk. Note that some lube oil scenarios appear to have been considered by PSL. Specifically, MFW and turbine lube oil fires were postulated. In speaking with the analysts, they indicated that other pumps tend not to have large quantities of lube oil and that source-target data for oil scenarios was often collected during walkdowns. However, there was little documentation of this, and very few oil scenarios were quantified in FRANC. PSL did not postulate H2 fires and oil fires as specified by NUREG/CR-6850, and minimal basis for this deviation was provided. These fires can be risk significant due to the potential for widespread damage in the fire compartment.	finding.
FSS-H1-01	FSS-H1	Not Met	In several cases, PSL implemented methods beyond those available in beyond industry accepted guidance documents (e.g., NUREG/CR-6850 and its supplements). For example, PSL created their own multipliers / severity factors for fires that cause damage beyond the ignition source by reviewing the EPRI Fire Events Database. A second example is that PSL modeled transient fires using the motor fire heat release rate distribution, which is much smaller than the transient fire distribution. A third example is not applying the "Location Factor" to account for wall/corner effects on flame height and plume temperature distribution. While these methods seem appropriate, documentation of the technical bases for these methods was generally lacking. Methods beyond industry accepted guidance (e.g., NUREG/CR-6850 and its supplements) should have documented technical bases of similar	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
			quality and magnitude to those provided in NUREG/CR-6850. Also, PSL should be aware that methods beyond industry accepted guidance documents may be viewed critically by the NRC. While these methods seem appropriate, the level of documentation provided did not allow detailed review by the peer reviewers. In addition, methods beyond industry accepted guidance (e.g., NUREG/CR-6850 and its supplements) should have documented technical bases of similar quality and magnitude to those provided in NUREG/CR-6850.	
HRA-A4-01	HRA-A4	Not Met	A review of modeled actions is planned to be performed once draft procedures are generated from the Fire PRA. However, at present no such review has been performed except for a limited board walkthrough documented in Appendix C of the Human Failure Evaluation report.	Documentation updates are needed to close this finding. The documentation updates will not affect the results. No impact on PRA applications.
PRM-C1-01	PRM-C1	Not Met	Overall PRM documentation is sparse and doesn't provide the information addressed in the SRs associated with the HLRs described in the Category I, II, and III criteria of PRM-C1. In addition, the development of changes made in Tables D1 and D3 are not described (PRM-B9). Recommend a separate PRM report that documents in a structured and consistent way the requirements described in the PRM SRs.	Documentation updates are needed to close this finding. The documentation updates will not affect the results. No impact on PRA applications.

Attachment 4: External Hazards Screening

External Hazard	Screening Result		
	Screened? (Y/N)	Screening Criterion (Note a)	Comment
Aircraft Impact	Y	PS4	Screened based on low probability of aircraft crash and small target size of SR structures.
Avalanche	Y	C3	Excluded due to site topography that would not support snow buildup that would lead to an avalanche.
Biological Event – Animal Infestation	Y	C4, C5	Included implicitly in LOOP initiator. Slow developing with limited impact.
Biological Event – Aquatic Grown	Y	C1 C5	Organic Material in Water is a more credible scenario to cause intake blockage than normal aquatic growth. Slow developing hazard, can be detected and managed.
Biological Event – Organic Material in Water	Y	C5	Slow developing hazard, can be detected and managed.
Coastal Erosion	Y	C1	Excluded based on design of plant.
Drought	Y	C3	Excluded since the capacity of the Ultimate Heat Sink (UHS) is not impacted by drought.
External Flooding	Y	C1	Cat. I SSCs protect from external flood by either design to withstand effects, located at sufficient grade to preclude inoperability or housed within waterproof structures. Leakage into Cat. I structures prevented by waterproofing penetrations and interconnections. [UFSAR 3.4.1]

Attachment 4: External Hazards Screening

External Hazard	Screening Result		
	Screened? (Y/N)	**Screening Criterion (Note a)**	**Comment**
Extreme Wind or Tornado	Y	C1	This event is of equal damage potential to tornados for which the plant has been designed. UFSAR 3.3 addresses design for wind loading and differential pressure loading. UFSAR Table 3.5-10 states that a missile equivalent to a 4 in. x 12 in. x 12 ft long wood plank (200 lb.) traveling end-on at 322 fps, a 1"dia. X 3' steel rod (8 lb.) traveling end-on at 163 fps, a 6" dia. Sch. 40 x 15' pipe (284.5 lb.) traveling end-on at 116 fps, a 12" dia. Sch. 40 x 15' pipe (743.4 lb.) traveling end-on at 116 fps, a 13.5" dia. X 35 ft. wooden utility pole (1497 lb.) traveling end-on at 153 fps and a passenger auto (4000 lb.) flying through the air at 84 fps and at not more than 25 ft above ground with a contact area of 20 square feet generated by the design basis tornado was considered in the design of the PSL.
Fog	Y	C4	Fog and mist may increase the frequency of accidents involving aircraft, ships, or vehicles. This weather condition is included implicitly in the accident rate data for these Transportation Accidents.
Forest or Range Fire	Y	C1 C3 C4	Included implicitly in LOOP initiator. Forest & grass are somewhat distant from the plant with no immediate impact on equipment.
Frost	Y	C4	Included implicitly in weather-related LOOP.

Attachment 4: External Hazards Screening

External Hazard	Screening Result		
	Screened? (Y/N)	Screening Criterion (Note a)	Comment
Hail	Y	C1 C4	Building design for high wind and missiles is bounding. Included implicitly in weather-related LOOP initiator.
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	C4	Included as a contributor to external flooding hazard. No impact alone.
Hurricane	Y	C4	Included as a contributor to external flooding hazard and high wind hazard. No impact alone. [UFSAR 2.4.2.5, 2.4.5].
Ice Cover	Y	C3	Ice blockage causing flooding is not applicable to the site because of location (climate conditions).
Industrial or Military Facility Accident	Y	C3	There are no industrial or military facilities in the vicinity of St. Lucie Nuclear Plant which would cause: 1) pressure wave that would fail a SR structure, 2) sufficient ground vibration for relay chatter, 3) control room habitability issues, or 4) chemical release into the water sufficient to impact the UHS
Internal Flooding	N	None	Internal Flooding PRA documented in PRA Notebook.

Attachment 4: External Hazards Screening

External Hazard	Screening Result		
	Screened? (Y/N)	**Screening Criterion (Note a)**	**Comment**
Internal Fire	N	None	Internal Fire PRA documented in PRA Notebooks.
Landslide	Y	C3	Excluded due to site topography that would not support landslide of any significance. The terrain in the site areas is essentially flat. Excluded based on location of SW intake connections some 31 feet below MLW, not subject to undermining from underwater landslide. PSL has two independent water sources, the Atlantic Ocean and Big Mud Creek.
Lightning	Y	C4 C1	Included implicitly in weather-related LOOP. Excluded based on design basis protection from lightning.
Low Lake Level or River Stage	Y	C3	Excluded based on location of intake which is approximately 30 feet below the surface of the Atlantic Ocean.
Low Winter Temperature	Y	C3	Excluded based on location of plant. Lowest recorded temperature 27 F in West Palm Beach.
Meteorite or Satellite Impact	Y	PS4	Conservative bounding assessment shows that these events can be screened. Extremely unlikely for satellite debris of any significant size to hit the site.
Pipeline Accident	Y	C3	There are no pipelines in the vicinity of St. Lucie Nuclear Plant.

Attachment 4: External Hazards Screening

External Hazard	Screening Result		
	Screened? (Y/N)	Screening Criterion (Note a)	Comment
Release of Chemicals in Onsite Storage	Y	C1	Control Room envelope is designed for habitability during normal, emergency and post-accident conditions.
River Diversion	Y	C3	Excluded since UHS does not depend on a river.
Sand or Dust Storm	Y	C1 C3	Plant equipment is protected from or designed to preclude foreign material. Excluded due to lack of large quantities of loose sand on site or nearby.
Seiche	Y	C3	Seiche is not considered because the site is facing the open coast on the east side and the Indian River on the west side.
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	C4	Included as a contributor to external flooding hazard. No impact alone.

Attachment 4: External Hazards Screening

External Hazard	Screening Result		
	Screened? (Y/N)	Screening Criterion (Note a)	Comment
Toxic Gas	Y	C1 PS4	Excluded based on design and low probability. Control Room envelope is designed for habitability during normal, emergency and post-accident conditions.
Transportation Accident	Y	C1 C3 C4	Conservative bounding assessment shows that these events can be screened.
Tsunami	Y	C1	The magnitude of such tsunami effects at the site are believed to be negligible compared to the effect of surges caused by the Probable Maximum Hurricane
Turbine-Generated Missiles	Y	C1 PS3	FPL complies with the turbine vendor's NRC approved inspection schedule and refurbishment recommendations. Mean frequency 1.88E-6/yr.
Volcanic Activity	Y	C3 C5	Excluded due to distance from nearest potentially active volcano. Any impact from a high elevation dust cloud would have long warning time.
Waves	Y	C4	Included as a contributor to external flooding hazard. No impact alone.
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.