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10 CFR 50.90
10 CFR 50.69

W3F1-2020-0047

December 18, 2020

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

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

Waterford Steam Electric Station, Unit 3
NRC Docket No. 50-382
Renewed Facility Operating License No. NPF-38

In accordance with the provisions of Sections 50.90 and 50.69 of Title 10 of the Code of Federal Regulations (10 CFR), Entergy Operations, Inc. (Entergy) is submitting a request for an amendment to the technical specifications (TS) for Waterford Steam Electric Station Unit 3 (Waterford 3).

The proposed amendment would modify the licensing basis, by the addition of a License Condition, to allow for the implementation of the provisions of Title 10 of the Code of Federal Regulations (10 CFR), Part 50.69, "Risk-Informed Categorization and Treatment of Structures, Systems, and Components (SSCs) for Nuclear Power Plants." The provisions of 10 CFR 50.69 allow adjustment of the scope of equipment subject to special treatment controls (e.g., quality assurance, testing, inspection, condition monitoring, assessment, and evaluation). For equipment determined to be of low safety significance, alternative treatment requirements can be implemented in accordance with this regulation. For equipment determined to be of high safety significance, requirements will not be changed or will be enhanced. This allows improved focus on equipment that has safety significance resulting in improved plant safety.

The enclosure to this letter provides the basis for the proposed change to the Waterford 3 Operating License. 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.

The NRC has previously reviewed the technical adequacy of the Waterford 3 Probabilistic Risk Assessment (PRA) model for both the adoption of TSTF-425, "Relocate Surveillance Frequencies to Licensee Control – RITSTF Initiative 5b" and the transition to NFPA-805, "Performance-Based Standard for Fire Protection for Light Water Reactor Generation Plants (2001 Edition)." Since then the PRA model has undergone updates with focused scoped peer reviews to close many of the open items. There are no PRA upgrades that have not been peer reviewed. Attachment 3 to the Enclosure provides additional detail.

Entergy requests approval of the proposed license amendment by January 30, 2021. The proposed changes would be implemented within 120 days of issuance of the amendment.

In accordance with 10 CFR 50.91(a)(1), "Notice for Public Comment," the analysis concerning the issue of no significant hazards consideration using the standards in 10 CFR 50.92 is included in Enclosure 1.

This letter contains no new regulatory commitments.

Should you have any questions or require additional information, please contact Paul Wood, Regulatory Assurance Manager, at 504-464-3786.

In accordance with 10 CFR 50.91, "Notice for public comment; State consultation," paragraph (b), a copy of this application, with attachments, is being provided to the designated State of Louisiana Official.

I declare under penalty of perjury, the foregoing is true and correct.
Executed on December 18, 2020.

Respectfully,

Ron Gaston

RWG/ajh

Enclosure: Evaluation of the Proposed Change

Attachments to Enclosure:

1. List of Categorization Prerequisites
2. Description of PRA Models Used in Categorization
3. Disposition and Resolution of Open Peer Review Findings and Self-Assessment Open Items
4. External Hazards Screening
5. Progressive Screening Approach for Addressing External Hazards
6. Disposition of Key Assumptions/Sources of Uncertainty

cc: NRC Region IV Regional Administrator
NRC Senior Resident Inspector – Waterford 3
Louisiana Department of Environmental Quality
NRC Project Manager – Waterford 3

# Enclosure

# W3F1-2020-0047

# Evaluation of the Proposed Change

**EVALUATION OF THE PROPOSED CHANGE**

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# EVALUATION OF THE PROPOSED CHANGE

## 1.0 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.

## 2.0 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 risk-informed 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 risk-informed 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 Waterford 3 to improve focus on equipment that has safety significance resulting in improved plant safety.

## 2.3 Description of the Proposed Change

Entergy proposes the addition of the following condition to the operating license of Waterford 3 to document the NRC's approval of the use 10 CFR 50.69:

> Entergy 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) using: Probabilistic Risk Assessment (PRA) models to evaluate risk associated with internal events, including internal flooding, and internal fire; the shutdown safety assessment process to assess shutdown risk; the Arkansas Nuclear One, Unit 2 (ANO-2) passive categorization method to assess passive component risk for Class 2 and Class 3 SSCs and their associated supports. The evaluation of impact of the seismic hazard uses the approach for seismic Tier 1 sites documented in Electric Power Research Institute (EPRI) Technical Update 3002017583, "Alternative Approaches for Addressing Seismic Risk in 10 CFR 50.69 Risk-Informed Categorization," which includes Waterford 3. Entergy/Waterford 3 will use a screening of other external hazards updated using the external hazard screening significance process identified in ASME/ANS PRA Standard RA-Sa-2009; as specified in License Amendment No. [XXX] dated [DATE].
>
> Prior NRC approval, under 10 CFR 50.90, is required for a change to the categorization process specified above (e.g., change from a seismic margins approach to a seismic probabilistic risk assessment approach).

## 3.0 TECHNICAL EVALUATION

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

> A licensee voluntarily choosing to implement this section shall submit an application for license amendment under § 50.90 that contains the following information:
>
> (i) A description of the process for categorization of RISC-1, RISC-2, RISC-3 and RISC-4 SSCs.
>
> (ii) A description of the measures taken to assure that the quality and level of detail of the systematic processes that evaluate the plant for internal and external events during normal operation, low power, and shutdown (including the plant-specific probabilistic risk assessment (PRA), margins-type approaches, or other systematic evaluation techniques used to evaluate severe accident vulnerabilities) are adequate for the categorization of SSCs.
>
> (iii) Results of the PRA review process conducted to meet § 50.69(c)(1)(i).

(iv) A description of, and basis for acceptability of, the evaluations to be conducted to satisfy § 50.69(c)(1)(iv). The evaluations must include the effects of common cause interaction susceptibility, and the potential impacts from known degradation mechanisms for both active and passive functions, and address internally and externally initiated events and plant operating modes (e.g., full power and shutdown conditions).

Each of these submittal requirements are addressed in the following sections.

## 3.1 Categorization Process Description (10 CFR 50.69(b)(2)(i))

### 3.1.1 Overall Categorization Process

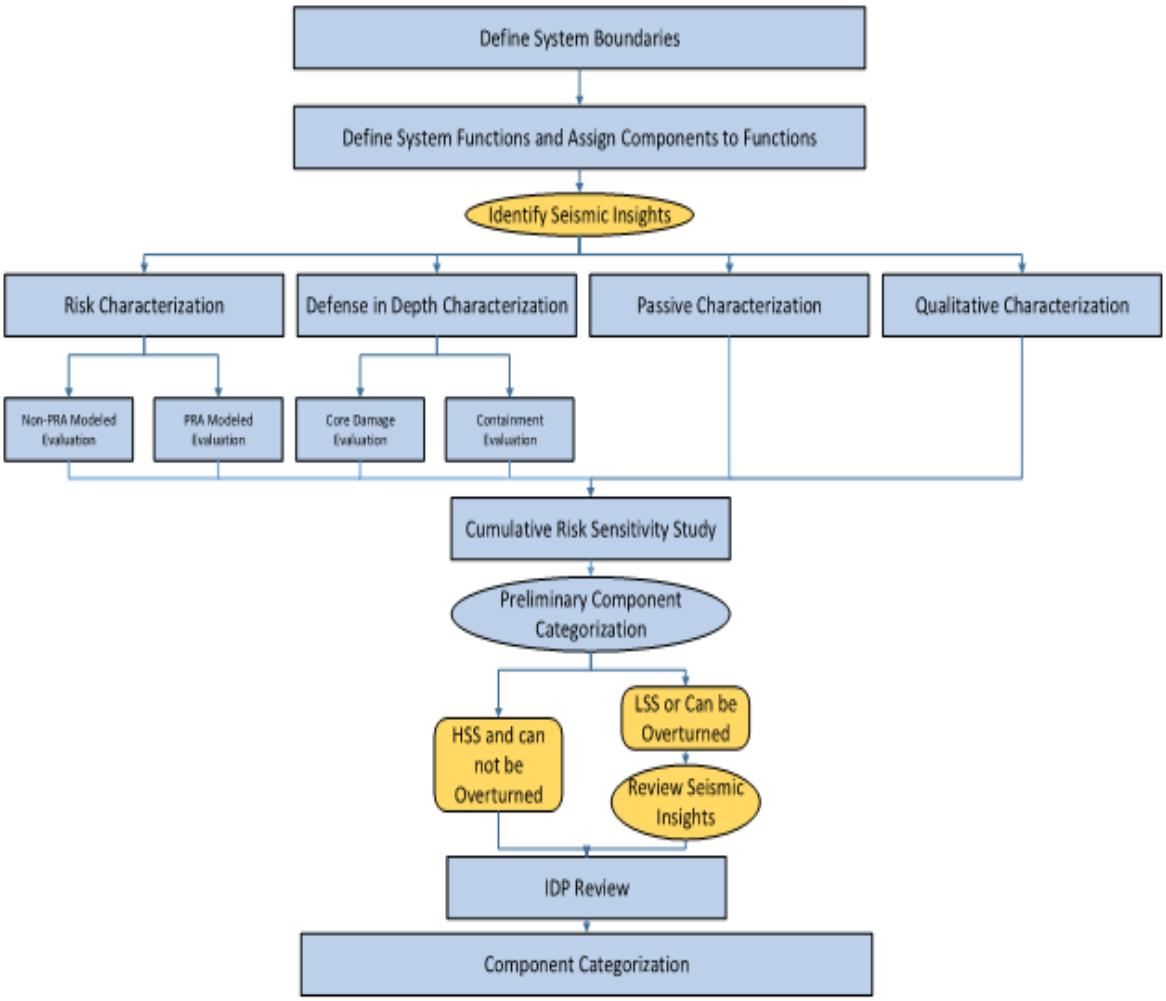
Waterford 3 will implement the risk categorization process in accordance with the NEI 00-04, Revision 0, as endorsed by Regulatory Guide (RG) 1.201, "Guidelines for Categorizing Structures, Systems, and Components in Nuclear Power Plants According to their Safety Significance," (Reference 2). NEI 00-04 Section 1.5 states, "Due to the varying levels of uncertainty and degrees of conservatism in the spectrum of risk contributors, the risk significance of SSCs is assessed separately from each of five risk perspectives and used to identify SSCs that are potentially safety-significant." A separate evaluation is appropriate to avoid reliance on a combined result that may mask the results of individual risk contributors.

The process to categorize each system will be consistent with the guidance in NEI 00-04, "10 CFR 50.69 SSC Categorization Guideline," as endorsed by RG 1.201, with the exception of the evaluation of impact of the seismic hazard, which will use the EPRI 3002017583 (Reference 3) approach for seismic Tier 1 sites, which includes Waterford 3, to assess seismic hazard risk for 10 CFR 50.69. Inclusion of additional process steps discussed below to address seismic considerations will ensure that reasonable confidence in the evaluations required by 10 CFR 50.69(c)(1)(iv) is achieved. RG 1.201 states that "the implementation of all processes described in NEI 00-04 (i.e., Sections 2 through 12) is integral to providing reasonable confidence" and that "all aspects of NEI 00-04 must be followed to achieve reasonable confidence in the evaluations required by § 50.69(c)(l)(iv)." However, neither RG 1.201 nor NEI 00-04 prescribe a particular sequence or order for each of the elements to be completed. Therefore, the order in which each of the elements of the categorization process (listed below) is completed is flexible and as long as they are all completed, they may even be performed in parallel. Note that NEI 00-04 only requires Item 3 to be completed for components/functions categorized as Low Safety Significant (LSS) by all other elements. Similarly, NEI 00-04 only requires Item 4 to be completed for safety-related active components/functions categorized as LSS by all other elements.

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

Figure 3-1 is an example of the major steps of the categorization process described in NEI 00-04; two steps (represented by four blocks on the figure) have been included to highlight review of seismic insights as pertains to this application, as explained further in Section 3.2.3:

Figure 3-1: Categorization Process Overview



Categorization of SSCs will be completed per the NEI 00-04 process, as endorsed by RG 1.201, which includes the determination of safety significance through the various elements identified above. The results of these elements are used as inputs to arrive at a preliminary component categorization (i.e., High Safety Significant (HSS) or LSS) that is presented to the Integrated Decision-Making Panel (IDP). Note: the term "preliminary HSS or LSS" is synonymous with the NEI 00-04 term "candidate HSS or LSS." A component or function is preliminarily categorized as HSS if any element of the process results in a preliminary HSS determination in accordance with Table 3-1 below. The safety significance determination of each element, identified above, is independent of each other and therefore the sequence of the elements does not impact the resulting preliminary categorization of each component or function. Consistent with NEI 00-04, the categorization of a component or function will only be "preliminary" until it has been confirmed by the IDP. Once the IDP confirms that the categorization process was followed appropriately, the final RISC category can be assigned.

The IDP may direct and approve detailed categorization of components in accordance with NEI 00-04 Section 10.2. The IDP may always elect to change a preliminary LSS component or function to HSS; however, the ability to change component categorization from preliminary HSS to LSS is limited. This ability is only available to the IDP for select process steps as described in NEI 00-04 and endorsed by RG 1.201. Table 3-1 summarizes these IDP limitations in NEI 00-04. The steps of the process are performed at either the function level, component level, or both. This is also summarized in the Table 3-1. A component is assigned its final RISC category upon approval by the IDP.

**Table 3-1: Categorization Evaluation Summary**

| Element | Categorization Step – NEI 00-04 Section | Evaluation Level | IDP Change HSS to LSS | Drives Associated Functions |
|---|---|---|---|---|
| Risk (PRA Modeled) | Internal Events Base Case – Section 5.1 | Component | Not Allowed | Yes |
| | Fire, Seismic and Other External Events Base Case | | Allowable | No |
| | PRA Sensitivity Studies | | Allowable | No |
| | Integral PRA Assessment – Section 5.6 | | Not Allowed | Yes |
| Risk (Non-modeled) | Fire, and Other External Hazards | Component | Not Allowed | No |
| | Seismic | Function/Component | Allowed[2] | No |
| | Shutdown – Section 5.5 | Function/Component | Not Allowed | No |
| Defense-in-Depth | Core Damage – Section 6.1 | Function/Component | Not Allowed | Yes |
| | Containment – Section 6.2 | Component | Not Allowed | Yes |
| Qualitative Criteria | Considerations – Section 9.2 | Function | Allowable[1] | N/A |
| Passive | Passive – Section 4 | Segment/Component | Not Allowed | No |

*Notes:*
*[1] The assessments of the qualitative considerations are agreed upon by the IDP in accordance with Section 9.2. In some cases, a 10 CFR 50.69 categorization team may provide preliminary assessments of the seven considerations for the IDP's consideration, however the final assessments of the seven considerations are the direct responsibility of the IDP.*

*The seven considerations are addressed preliminarily by the 10 CFR 50.69 categorization team for at least the system functions that are not found to be HSS due to any other categorization step. Each of the seven considerations requires a supporting justification for confirming (true response) or not confirming (false response) that consideration. If the 10 CFR 50.69 categorization team determines that one or more of the seven considerations cannot be confirmed, then that function is presented to the IDP as*

*preliminary HSS. Conversely, if all the seven considerations are confirmed, then the function is presented to the IDP as preliminary LSS.*

*The System Categorization Document, including the justifications provided for the qualitative considerations, is reviewed by the IDP. The IDP is responsible for reviewing the preliminary assessment to the same level of detail as the 10 CFR 50.69 team (i.e. all considerations for all functions are reviewed). The IDP may confirm the preliminary function risk and associated justification or may direct that it be changed based upon their expert knowledge. Because the Qualitative Criteria are the direct responsibility of the IDP, changes may be made from preliminary HSS to LSS or from preliminary LSS to HSS at the discretion of the IDP. If the IDP determines any of the seven considerations cannot be confirmed (false response) for a function, then the final categorization of that function is HSS.*

[2] *IDP consideration of seismic insights can also result in an LSS to HSS determination.*

The mapping of components to system functions is used in some categorization process steps to facilitate preliminary categorization of components. Specifically, functions with mapped components that are determined to be HSS by the PRA-based assessment (i.e., Internal events PRA or Integral PRA assessment) or defense-in-depth evaluation will be initially treated as HSS. However, NEI 00-04 Section 10.2 allows detailed categorization which can result in some components mapped to HSS functions being treated as LSS; and Section 4.0 discusses additional functions that may be identified (e.g., fill and drain) to group and consider potentially LSS components that may have been initially associated with a HSS function but which do not support the critical attributes of that HSS function. Note that certain steps of the categorization process are performed at a component level (e.g. Passive, Non-PRA-modeled hazards – see Table 3-1). Except for seismic, these components from the component level assessments will remain HSS (IDP cannot override) regardless of the significance of the functions to which they are mapped. Components having seismic functions may be HSS or LSS based on the IDP's consideration of the seismic insights applicable to the system being categorized. Therefore, if a HSS component is mapped to an LSS function, that component will remain HSS. If an LSS component is mapped to an HSS function, that component may be driven HSS based on Table 3-1 above or may remain LSS. For the seismic hazard, given that Waterford 3 is a seismic Tier 1 (low seismic hazard) plant as defined in Reference 3, seismic considerations are not required to drive an HSS determination at the component level, but the IDP will consider available seismic information pertinent to the components being categorized and can, at its discretion, determine that a component should be HSS based on that information.

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

- The 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 DBEs; 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 10 CFR 50.69(f)(1) will be documented in Entergy procedures. Decisions of the IDP will be arrived at by consensus. Differing opinions will be documented and resolved, if possible. However, a simple majority of the panel is sufficient for final decisions regarding HSS and LSS.

- Passive characterization will be performed using the processes described in Section 3.1.2. Consistent with NEI 00-04, an HSS determination by the passive categorization process cannot be changed by the IDP.

- An unreliability factor of 3 will be used for the risk evaluation 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.

- NEI 00-04 Section 7 requires assigning the safety significance of functions to be preliminary HSS if it is supported by an SSC determined to be HSS from the PRA-based assessment in Section 5, but does not require this for SSCs determined to be HSS from non-PRA-based, deterministic assessments in Section 5. This requirement is further clarified in the Vogtle SE (Reference 4) which states "…if any SSC is identified as HSS from either the integrated PRA component safety significance assessment (Section 5 of NEI 00-04) or the defense-in-depth assessment (Section 6), the associated system function(s) would be identified as HSS."

- 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 LSS.

- With regard to the criteria that considers whether the active function is called out or relied upon in the plant Emergency/Off-Normal Operating Procedures, Waterford 3 will not take credit for alternate means unless the alternate means are proceduralized and included in Licensed Operator training.

- Waterford 3 proposes to apply an alternative seismic approach to those listed in NEI 00-04 Sections 1.5 and 5.3. This approach is specified in EPRI 3002017583 (Reference 3) for Tier 1 plants and is discussed in Section 3.2.3.

The risk analysis to be implemented for each modeled hazard is described below.

- Internal Event Risks: Internal events including internal flooding PRA model Revision 6 (refer to Attachment 2).

- Fire Risks: Fire PRA model Revision 6 (refer to Attachment 2).

- Seismic Risks: EPRI Alternative Approach in EPRI 3002017583 (Reference 3) for Tier 1 plants with the additional considerations discussed in Section 3.2.3 of this LAR.

- Other External Risks (e.g., tornados, external floods, etc.): These were determined to be insignificant contributors to plant risk. A 2017 analysis (PSA-WF3-07-01 - Waterford 3 Re-Examination of External Events Evaluation in the IPEEE - Reference 26) was completed to re-examine IPEEE External Events to ensure the IPEEE conclusions are

valid and to account for updated events/consequence data and plant design. This evaluation of external hazards was performed using Part 6 of the ASME/ANS PRA Standard (Reference 5).

- 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 6), 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 7 consistent with the related Safety Evaluation (SE) 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., DID, 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. Consistent with NEI 00-04, an HSS determination by the passive categorization process cannot be changed by the IDP. The use of this method was previously approved to be used for a 10 CFR 50.69 application by NRC in the final SE for Vogtle dated December 17, 2014 (Reference 4). 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. The passive categorization process is intended to apply the same risk-informed process accepted by the NRC in the ANO2-R&R-004 for the passive categorization of Class 2, 3, and non-class components. This is the same passive SSC scope the NRC has conditionally endorsed in ASME Code Cases N-660 and N-662-1 as published in RG 1.147, Revision 19 (Reference 17). Both code cases employ a similar risk-informed safety classification of SSCs in order to change the repair/replacement requirements of the affected LSS components. All ASME Code Class 1 SSCs with a pressure retaining function, as well as supports, will be assigned high safety-significant, HSS, for passive categorization which will result in HSS for its risk-informed safety classification and cannot be changed by the IDP. Therefore, this methodology and scope for passive categorization is acceptable and appropriate for use at Waterford 3 for 10 CFR 50.69 SSC categorization.

## 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. The baseline internal events PRA and Fire PRA models credited in this request are the same PRA models credited in the TSTF-425 and NFPA 805 approved license amendments (References 46 and 47, respectively) with routine maintenance updates applied. The internal flood model has been revised with method upgrades, and was peer reviewed upon completion. The PRA models described below have been peer reviewed and there are no PRA upgrades that have not been peer reviewed.

### 3.2.1 Internal Events and Internal Flooding

The Waterford 3 categorization process for the internal events and flooding hazard will use the plant-specific PRA model. The Entergy risk management process ensures that the PRA model used in this application reflects the as-built and as-operated plant for Waterford 3. Attachment 2 of this enclosure identifies the applicable internal events and internal flooding PRA models.

### 3.2.2 Internal Fire Hazards

The Waterford 3 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 (Reference 25) and only utilizes methods previously accepted by the NRC. The Entergy risk management process ensures that the PRA model used in this application reflects the as-built and as-operated plant for Waterford 3. Attachment 2 of this enclosure identifies the applicable Fire PRA model.

### 3.2.3 Seismic Hazards

10 CFR 50.69(c)(I) requires the use of PRA to assess risk from internal events. For other risk hazards such as seismic, 10 CFR 50.69 (b)(2) allows, and NEI 00-04 summarizes, the use of other methods for determining SSC functional importance in the absence of a quantifiable PRA (such as Seismic Margin Analysis or IPEEE Screening) as part of an integrated, systematic process. For the Waterford 3 seismic hazard assessment, Entergy proposes to use a risk informed graded approach that meets the requirements of 10 CFR 50.69 (b)(2) as an alternative to those listed in NEI 00-04 sections 1.5 and 5.3. This approach is outlined in Electric Power

Research Institute (EPRI) report 3002017583 (Reference 3), and includes additional qualitative considerations that are discussed in this section.

Waterford 3 meets the Tier 1 criteria for a "Low Seismic Hazard/High Seismic Margin" site. The Tier 1 criteria are as follows:

> Tier 1: Plants where the GMRS [Ground Motion Response Spectrum] peak acceleration is at or below approximately 0.2g or where the GMRS is below or approximately equal to the SSE (Safe Shutdown Earthquake) between 1.0 Hz and 10 Hz. Examples are shown in Figures 2-1 and 2-2 of the EPRI report (Reference 3). At these sites, the GMRS is either very low or within the range of the SSE such that unique seismic categorization insights are not expected.

*Note:* Reference 3 applies to the Tier 1 sites in its entirety except for the sections 2.3 (Tier 2 sites), 2.4 (Tier 3 sites), Appendix A (seismic correlation), and Appendix B (criteria for capacity-based screening).

The Tier 1 criterion (i.e. basis) in the EPRI report (Reference 3) is a comparison of the ground motion response spectrum (GMRS, derived from the seismic hazard) to the safe shutdown earthquake (SSE, i.e., seismic design basis capability). U.S. nuclear power plants that utilize this approach will continue to compare GMRS to SSE.

The trial studies presented in the EPRI report (Reference 3) show that seismic categorization insights are overlaid by other risk insights even at plants where the GMRS is far beyond the seismic design basis. Therefore, the basis for the Tier 1 classification and resulting criteria is not that the design basis insights are adequate. Instead, it is that consideration of the full range of the seismic hazard produces limited unique insights to the categorization process. That is the basis for the following statements in Table 4-1 of the EPRI report.

> *At Tier 1 sites, the likelihood of identifying a unique seismic condition that would cause an SSC to be designated HSS is very low.*
>
> *Therefore, with little to no anticipated unique seismic insights, the 50.69 categorization process using the FPIE [Full Power Internal Events] PRA and other risk evaluations along with the required Defense-in-Depth and IDP qualitative considerations are expected to adequately identify the safety-significant functions and SSCs required for those functions and no additional seismic reviews are necessary for 50.69 categorization.*

The proposed categorization approach for Waterford 3 is a risk-informed graded approach that is demonstrated to produce categorization insights equivalent to a seismic PRA. For Tier 1 plants, this approach relies on the insights gained from the seismic PRAs examined in Reference 3 along with confirmation that the site GMRS is low. Reference 3 demonstrates that seismic risk is adequately addressed for Tier 1 sites by the results of additional qualitative assessments discussed in this section and existing elements of the 10 CFR 50.69 categorization process specified in NEI 00-04.

For example, the 10 CFR 50.69 categorization process as defined in NEI 00-04 includes an Integral Assessment that weighs the hazard-specific relative importance of a component (e.g., internal events, internal fire, seismic) by the fraction of the total Core Damage Frequency

(CDF) contributed by that hazard. For Tier 1 sites, the seismic risk (CDF/Large Early Release Frequency (LERF)) will be low such that seismic hazard risk is unlikely to influence an HSS decision. In applying the EPRI report (Reference 3) process for Tier 1 sites to the Waterford 3 10 CFR 50.69 categorization process, the IDP will be provided with the rationale for applying the EPRI guidance and informed of plant SSC-specific seismic insights for their consideration in the HSS/LSS deliberations.

The EPRI report (Reference 3) recommends a risk-informed graded approach for addressing the seismic hazard in the 10 CFR 50.69 categorization process. The guidance document contains several seismic fragility fundamental concepts that support a graded approach and includes important characteristics about the comparison of the seismic design basis (represented by the SSE) to the site-specific seismic hazard (represented by the GMRS) that support the selected thresholds between the three evaluation Tiers in the EPRI report. The coupling of these concepts with the categorization process in NEI 00-04 are the key elements of the approach defined in Reference 3 for identifying unique seismic insights.

The seismic fragility of an SSC is a function of the margin between an SSC's seismic capacity and the site-specific seismic demand. References such as EPRI NP-6041-SL (Reference 8) provide inherent seismic capacities for most SSCs that are not directly related to the site-specific seismic demand. This inherent seismic capacity is based on the non-seismic design loads (pressure, thermal, dead weight, etc.) and the required functions for the SSC. For example, a pump has a relatively high inherent seismic capacity based on its design and that same seismic capacity applies at a site with a very low demand and at a site with a very high demand. At sites with lower seismic demands such as Waterford 3, there is no need to perform more detailed evaluations to demonstrate the inherent seismic capacities documented in industry sources such as Reference 8. Low seismic demand sites have lower likelihood of seismically-induced failures and lesser challenges to plant systems. This, therefore, provides the technical basis for allowing use of a graded approach for addressing seismic hazards at Waterford 3.

There are some plant features such as equipment anchorage that have seismic capacities more closely associated with the site-specific seismic demand since those specific features are specifically designed to meet that demand. However, even for these features, the design basis criteria have intended conservatisms that result in significant seismic margins within SSCs. These conservatisms are reflected in key aspects of the seismic design process. The SSCs used in nuclear power plants are intentionally designed using conservative methods and criteria to ensure that they have margins well above the required design bases. Experience has shown that design practices result in margins to realistic seismic capacities of 1.5 or more.

The following provides the basis for establishing Tier 1 criteria in the EPRI report (Reference 3):

a. SSCs for which the inherent seismic capacities are applicable, or which are designed to the plant SSE, will have low probabilities of failure at sites where the peak spectral acceleration of the GMRS < 0.2g or where the GMRS < SSE between 1 and 10 Hz.

b. The low probabilities of failure of individual components would also apply to components considered to have correlated seismic failures.

c. These low probabilities of failure lead to low seismic CDF and LERF estimates, from an absolute risk perspective.

d. The low seismic CDF and LERF estimates lead to reasonable confidence that seismic risk contributions would allow reducing an HSS to LSS due to the 10 CFR 50.69 Integral Assessment if the equipment is HSS only due to seismic considerations.

Test cases described in Section 3 of the EPRI report (Reference 3) showed that it would be unusual even for moderate hazard plants to exhibit any unique seismic insights, including due to correlated failures. Hence, while it is prudent to perform additional evaluations to identify conditions where correlated failures may occur for Tier 2 sites, for Tier 1 sites such as Waterford 3, correlation studies would not lead to new seismic insights or affect the site risk results in any significant way.

The Tier 1 to Tier 2 threshold as defined in Reference 3 provides a clear and traceable boundary that can be consistently applied plant site to plant site. Additionally, because the boundary is well defined, if new information is obtained on the site hazard, a site's location within a particular Tier can be readily confirmed. In the unlikely event that the Waterford 3 seismic hazard changes to medium risk (i.e., Tier 2) at some future time, Waterford 3 will follow its categorization review and adjustment process procedures to review the changes to the plant and update, as appropriate, the SSC categorization in accordance with 10 CFR 50.69(e).

The following provides the basis for concluding that Waterford 3 meets the Tier 1 site criteria.

In response to the NRC 50.54(f) letter associated with post-Fukushima recommendations (Reference 10), Waterford 3 submitted a seismic hazard screening report (Reference 11) to the NRC. The maximum GMRS value for Waterford 3 1-10 Hz range meets the Tier 1 criterion of approximately 0.2g provided in Reference 3.

Both the Waterford 3 and NRC generated plots for SSE and GMRS curves from the seismic hazard and screening are plotted in Figure A4-1 at the end of Attachment 4. The NRC's staff assessment of the Waterford 3 seismic hazard and screening response is documented in Reference 13. In section 3.4 of Reference 13, the NRC concluded that the methodology used by Entergy in determining the GMRS was acceptable and that the GMRS determined by Entergy adequately characterizes the reevaluated hazard for the Waterford 3 site.

Section 1.1.3 of Reference 3 cites various post-Fukushima seismic reviews performed for the U.S. fleet of nuclear power plants. For Waterford 3, the specific seismic reviews prepared by the licensee and the NRC's staff assessments are provided here. These licensee documents were submitted under oath and affirmation to the NRC.

1) NTTF Recommendation 2.1 seismic hazard screening (Reference 11)
2) NTTF Recommendation 2.3 seismic walkdowns (Reference 14)

In addition to the NTTF related seismic studies, Waterford 3 completed a bounding seismic risk evaluation (Reference 27) to support development of a Risk-informed Completion Time (TSTF-505) License Amendment Request and program. A seismic penalty based on the total estimated seismic risk will be applied to Risk-Informed Completion Time (RICT) evaluations. The resulting (overstated) seismic penalty that will be applied to RICT program is approximately

3E-6/yr for CDF . This value is an order of magnitude below the summed-up CDF total of the detailed PRA models for the site (2.56E-5/yr listed in Attachment 2). That seismic risk estimate will not be used in 10 CFR 50.69 related categorization but can be used to inform the IDP of the maximum potential contribution from seismic risk.

As an enhancement to the EPRI study results as they pertain to Waterford 3, the proposed Waterford 3 categorization approach for seismic hazards will include qualitative consideration of the mitigation capabilities of SSCs during seismically-induced events and seismic failure modes, based on insights obtained from prior seismic evaluations performed for Waterford 3. For example, as part of the categorization team's preparation of the System Categorization Document (SCD) that is presented to the IDP, a section will be included in the SCD that summarizes the identified plant seismic insights pertinent to the system being categorized and will also state the basis for applicability of the EPRI study (Reference 3)and the bases for Waterford 3 being a Tier 1 plant. The discussion of the Tier 1 bases will include such factors as:

- The low seismic hazard for the plant, which is subject to periodic reconsideration as new information becomes available through industry evaluations; and

- The definition of Tier 1 in the EPRI study.

  At several steps of the categorization process (e.g., as noted in Figure 3-1 and Table 3-1 of Reference 3) the categorization team will consider the available seismic insights relative to the system being categorized and document their conclusions in the SCD. Integrated importance measures over all modeled hazards (i.e., internal events, including internal flooding, and internal fire for Waterford 3) are calculated per Section 5.6 of NEI 00-04, and components for which these measures exceed the specified criteria are preliminary HSS which cannot be changed to LSS. All SCCs in the systems subject to categorization, that have design-basis functions during seismic events or functions credited for mitigation and prevention of severe accidents caused by seismic events, will be addressed using non-PRA based qualitative assessments in conjunction with any seismic insights provided by the PRA.

For components that are HSS due to fire PRA but not HSS due to internal events PRA, the categorization team will review design-basis functions during seismic events or functions credited for mitigation and prevention of severe accidents caused by seismic events and characterize these for presentation to the IDP as additional qualitative inputs, which will also be described in the SCD.

The categorization team will review available Waterford 3 plant-specific seismic reviews and other resources such as those identified above. The objective is to identify plant-specific seismic insights derived from the above sources, relevant to the components in the system being categorized, that might include potentially important impacts such as:

- Impact of relay chatter
- Implications related to potential seismic interactions such as with block walls
- Seismic failures of passive SSCs such as tanks and heat exchangers
- Any known structural or anchorage issues with a particular SSC
- Components that are implicitly part of PRA-modeled functions (including relays)
- Components that may be subject to correlated failures

Such impacts would be compiled on an SSC basis. As each system is categorized, the system specific seismic insights will be provided to the IDP for consideration as part of the IDP review process, as noted in Figure 3-1 of the EPRI report (Reference 3). As such, the IDP can challenge, from a seismic perspective, any candidate LSS recommendation for any SSC if they believe there is basis for doing so. Any decision by the IDP to lower preliminary HSS components to LSS will also consider the applicable seismic insights in that decision. These insights will provide the IDP a means to consider potential impacts of seismic events in the categorization process.

Use of the EPRI approach outlined in Reference 3 to assess seismic hazard risk for 10 CFR 50.69 with the additional reviews discussed above will provide a process for categorization of RISC-1, RISC-2, RISC-3, and RISC-4 SSCs that satisfies the requirements of 10 CFR 50.69(c).

Based on the above, Section 2.2.3 of the EPRI report (Reference 3) applies to Waterford 3, i.e., Waterford 3 is a Tier 1 plant for which the GMRS is very low such that unique seismic categorization insights are expected to be minimal. As discussed in Reference 3, the likelihood of identifying a unique seismic insight that would cause an SSC to be designated HSS is very low. Therefore, with little to no anticipated unique seismic insights, the 10 CFR 50.69 categorization process using the Full Power Internal Events (FPIE) PRA and other risk evaluations along with the defense-in-depth and qualitative assessment by the IDP adequately identify the safety-significant functions and SSCs.

### 3.2.4 Other External Hazards

All other external hazards, except for seismic, were screened for applicability to Waterford 3 per a plant-specific evaluation in accordance with Generic Letter (GL) 88-20 (Reference 15) and updated to use the criteria in ASME PRA Standard RA-Sa-2009 (Reference 5). A 2017 analysis was completed to re-examine the IPEEE external hazards, PRA-WF3-07-01 (Reference 26). This effort included re-examining the hazards to the current design basis and as-built plant, and examining newer data for hazards. 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

Consistent with NEI 00-04, the Waterford 3 categorization process will use the shutdown safety management plan described in NUMARC 91-06 (Reference 6) for evaluation of safety significance related to low power and shutdown conditions. The overall process for addressing shutdown risk is illustrated in Figure 5-7 of NEI 00-04.

NUMARC 91-06 specifies that a defense-in-depth approach should be used with respect to each defined shutdown key safety function. The key safety functions defined in NUMARC 91-06 are evaluated for categorization of SSCs.

SSCs that meet either of the two criteria (i.e., considered part of a "primary shutdown safety system" or a failure would initiate an event during shutdown conditions) described in Section 5.5 of NEI 00-04 will be considered preliminary HSS.

### 3.2.6 PRA Maintenance and Updates

The Entergy risk management process ensures that the applicable PRA models used in this application continues to reflect the as-built and as-operated plant for Waterford 3. 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, and 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, Entergy 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

### 3.2.7 PRA Uncertainty Evaluations

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

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

In the overall risk evaluations, Entergy will utilize a factor of 3 to increase the unavailability or unreliability of LSS components. Consistent with the NEI 00-04 guidance, Entergy will perform both an initial risk evaluation and a cumulative evaluation study. The initial risk evaluation study applies to the system that is being categorized. In the cumulative risk evaluation 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 evaluation 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 risk evaluation.

The detailed process of identifying, characterizing and qualitative screening of model uncertainties is found in Section 5.3 of NUREG-1855 (Reference 21) 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 which would be significant for the evaluation of this application. If the Waterford 3 PRA model used a nonconservative 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 Waterford 3 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 Waterford 3 PRA model specific assumptions or sources of uncertainty. Note that NEI guidance requires sensitivity evaluations for common cause and human failure events.

### 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 12), consistent with NRC RIS 2007-06 (Reference 50). The Internal Events PRA model (including internal flood) was peer reviewed in 2009 by the PWR Owners Group (PWROG) prior to the issuance of Revision 2 of RG 1.200. A gap-assessment was conducted to assess the differences between RG 1.200 Rev 2 and RG 1.200 Rev 1. That assessment confirmed that the PRA model meets the requirements of RG 1.200 Rev 2. Results from that assessment are documented in Attachment 3. The Internal Events PRA technical adequacy (including the 2009 peer review and self-assessment results) has previously been reviewed by the NRC in previous requests associated with the NFPA 805 license amendment (Reference 47).

The Fire PRA model was subject to a self-assessment and a full-scope peer review conducted in February 2011. Following a 2012 NRC Audit of the Waterford 3 NFPA 805 LAR and supporting documents, Waterford 3 revised the Fire PRA including several method changes. Waterford 3 completed two focused scope peer reviews (September 2012 and May 2013) to ensure proper evaluation of the revised methods.

The results of those peer reviews (one for internal events and the full scope and two focused scope reviews for fire) were the basis for the NFPA 805 LAR and SE and the TSTF-425 program LAR and SE. Subsequently, the Waterford 3 PRA models have been through several updates (with some upgrades) as well as peer reviews and formal Finding and Observation Close-out reviews.

Findings for both the full power Internal Events and Fire PRA models (as of October 2017) were reviewed and closed using the process documented in Appendix X to NEI 05-04, NEI 07-12, and NEI 12-13, "Close-out of Facts and Observations (F&Os)," (Reference 19) as accepted by NRC in the staff memorandum dated May 3, 2017 (Reference 51). The results of this review have been documented and are available for NRC audit. The closure review concluded that the site response to one finding resulted in application of a modeling upgrade. This one PRA upgrade as defined by the ASME PRA Standard RA-Sa-2009 (Reference 5) has occurred to the Internal Events PRA model since conduct of the PWROG peer review in 2009. That upgrade is associated with cooling tower success criteria. It was reviewed during the F&O closure review in October 2017 (as an imbedded focused peer review).

Following the closure review, both the internal events model and fire PRA model were revised to address findings that were not closed during the closure review (in addition to a routine model update). During this update, the LERF model was revised with model upgrades to resolve

issues and more thoroughly meet the PRA Standard. The LERF model was subject to a peer review in August 2019 as a result of the upgraded methods used.

The Waterford 3 internal flood PRA model was also revised in 2019 and 2020. The flood PRA model update also included methodology upgrades. The updated flood model was subject to a peer review in August 2019 for all PRA Standard elements relevant to internal flooding PRA models. Following the August peer review, the model was revised to address the resulting Findings.

The above 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 by 10 CFR 50.69(c)(1)(i).

### 3.4 Risk Evaluations (10 CFR 50.69(b)(2)(iv))

The Waterford 3 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 10 CFR 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 CDF and 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, and human errors). 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.

### 3.5 Feedback and Adjustment Process

If significant changes to the plant risk profile are identified, or if it is identified that a RISC-3 or RISC-4 SSC can (or actually did) prevent a safety significant function from being satisfied, an immediate evaluation and review will be performed prior to the normally scheduled periodic review. Otherwise, the assessment of potential equipment performance changes and new technical information will be performed during the normally scheduled periodic review cycle. To more specifically address the feedback and adjustment (i.e., performance monitoring) process as it pertains to the proposed Waterford 3 Tier 1 approach discussed in section 3.2.3, implementation of the Waterford 3 design control and corrective action programs will ensure the inputs for the qualitative determinations for seismic continue to remain valid to maintain compliance with the requirements of 10 CFR 50.69(e).

The performance monitoring process will be described in the Waterford 3 10 CFR 50.69 program documents. The program requires that the periodic review assess changes that could impact the categorization results and provides the IDP with an opportunity to recommend categorization and treatment adjustments. Station personnel from engineering, operations, risk management, regulatory affairs, and others have responsibilities for preparing and conducting various performance monitoring tasks that feed into this process. The intent of the performance monitoring reviews is to discover trends in component reliability, to identify and reverse negative performance trends, and take corrective action if necessary. The Waterford 3 configuration control process ensures that changes to the plant, including a physical change to the plant and changes to documents, are evaluated to determine the impact to drawings, design bases,

licensing documents, programs, procedures, and training. The configuration control program will include a checklist of configuration activities to recognize those systems that have been categorized in accordance with 10 CFR 50.69 to ensure that any physical change to the plant or change to plant documents is evaluated prior to implementing those changes.

The checklist includes:

- A review of the impact on the System Categorization Document (SCD) for configuration changes that may impact a categorized system under 10 CFR 50.69.

- Steps to be performed if redundancy, diversity, or separation requirements are identified or affected. These steps include identifying any potential seismic interaction between added or modified components and new or existing safety related or safe shutdown components or structures. Review of impact to seismic loading, safe shutdown earthquake (SSE) seismic requirements, as well as the method of combining seismic components.

- Review of seismic dynamic qualification of components if the configuration change adds, relocates, or alters Seismic Category I mechanical or electrical components.

- Waterford 3 uses Entergy's comprehensive problem identification and corrective action program that ensures that issues are identified and resolved. Any issue that may impact the 10 CFR 50.69 categorization process will be identified and addressed through the problem identification and corrective action program, including seismic-related issues.

The Waterford 3 10 CFR 50.69 program requires that SCDs cannot be approved by the IDP until the panel's comments have been resolved to the satisfaction of the IDP. This includes issues related to system-specific seismic insights considered by the IDP during categorization. Scheduled periodic reviews at least once every other refueling outage will evaluate new insights resulting from available risk information (i.e., PRA model or other analysis used in the categorization) changes, design changes, operational changes, and SSC performance. If it is determined that these changes have affected the risk information or other elements of the categorization process such that the categorization results are more than minimally affected, then the risk information and the categorization process will be updated. This scheduled review will include:

- A review of plant modifications since the last review that could impact the SSC categorization.
- A review of plant specific operating experience that could impact the SSC categorization.
- A review of the impact of the updated risk information on the categorization process results.
- A review of the importance measures used for screening in the categorization process.
- An update of the risk sensitivity study performed for the categorization.

In addition to the normally scheduled periodic reviews, if a PRA model or other risk information is upgraded, a review of the SSC categorization will be performed.

The periodic monitoring requirements of the 10 CFR 50.69 process will ensure that these issues are captured and addressed at a frequency commensurate with the issue severity. The 10 CFR 50.69 periodic monitoring program includes immediate and periodic reviews, that

include the requirements of the regulation, to ensure that all issues that could affect 10 CFR 50.69 categorization are addressed. The periodic monitoring process also monitors the performance and condition of categorized SSCs to ensure that the assumptions for reliability in the categorization process are maintained.

## 4.0 REGULATORY EVALUATION

### 4.1 Applicable Regulatory Requirements/Criteria

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

- The regulations in 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 3, January 2018.

- Regulatory Guide 1.200, "An Approach for Determining the Technical Adequacy of Probabilistic Risk Assessment Results for Risk-Informed Activities," Revision 2, March 2009.

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

### 4.2 No Significant Hazards Consideration Analysis

Waterford 3 proposes to modify the licensing basis to allow for the voluntary implementation of the provisions of Title 10 of the Code of Federal Regulations (10 CFR), Part 50.69, "Risk-Informed Categorization and Treatment of Structures, Systems and Components 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.

Waterford 3 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 planned change will permit the use of a risk-informed categorization process to modify the scope of Structures, Systems and Components (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, Waterford 3 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.

## 5.0 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.

## 6.0 REFERENCES

1. Nuclear Energy Institute (NEI) Report NEI 00-04, "10 CFR 50.69 SSC Categorization Guideline," Revision 0, (ADAMS Accession No. ML052910035), dated July 2005

2. NRC Regulatory Guide 1.201, "Guidelines for Categorizing Structures, Systems, and Components in Nuclear Power Plants According to their Safety Significance," Revision 1, (ADAMS Accession No. ML061090627), dated May 2006

3. Electric Power Research Institute (EPRI) Technical Update 3002017583, "Alternative Approaches for Addressing Seismic Risk in 10 CFR 50.69 Risk-Informed Categorization," dated February 11, 2020

4. NRC letter to Southern Nuclear Operating Company, Inc., "Vogtle Electric Generating Plant, Units 1 and 2 - Issuance of Amendments Re: Use of 10 CFR 50.69 (TAC Nos. ME9472 and ME94473)," (ADAMS Accession No. ML14237A034), dated December 17, 2014

5. American Society of Mechanical Engineers/American Nuclear Society (ASME/ANS), "Standard for Level I/Large Early Release Frequency Probabilistic Risk Assessment for Nuclear Power Plant Applications," RA-Sa-2009, Addendum A to RA-S-2008, dated February 2009

6. Nuclear Management and Resources Council, Inc. (NUMARC) Report 91-06, "Guidelines for Industry Actions to Assess Shutdown Management," (ADAMS Accession No. ML14365A203), dated December 1991

7. NRC letter to Entergy Operations, Inc. (Entergy), "Arkansas Nuclear One, Unit 2 - Approval of Request for Alternative ANO2-R&R-004, Revision 1, Request to Use Risk-Informed Safety Classification and Treatment for Repair/Replacement Activities in Class 2 and 3 Moderate and High Energy Systems (TAC No. MD5250)," (ADAMS Accession No. ML090930246), dated April 22, 2009

8. EPRI Technical Report NP-6041-SL, "A Methodology for Assessment of Nuclear Plant Seismic Margin (Revision 1)," dated August 1, 1991

9. ASME, "Standard for Probabilistic Risk Assessment for Nuclear Power Plant Applications," RA-Sb-2005, Addendum B to ASME RA-S-2002, dated December 30, 2005

10. NRC letter to All Power Reactor Licensees and Holders of Construction Permits in Active or Deferred Status, "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," (ADAMS Accession No. ML12053A340), dated March 12, 2012

11. Entergy letter to NRC, "Entergy Seismic Hazard and Screening Report (CEUS Sites), Response to NRC Request for Information Pursuant to 10 CFR 50.54(f) Regarding Recommendation 2.1 of the Near-Term Task Force Review of Insights from the Fukushima Dai-ichi Accident," W3F1-2014-0023, (ADAMS Accession No. ML14086A427), dated March 27, 2014

12. NRC Regulatory Guide 1.200, "An Approach for Determining the Technical Adequacy of Probabilistic Risk Assessment Results for Risk-Informed Activities," Revision 2, (ADAMS Accession No. ML090410014), dated March 2009

13. NRC letter to Entergy, "Waterford Steam Electric Station, Unit 3 - Staff Assessment 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 (TAC No. MF3712)," (ADAMS Accession No. ML15335A050), dated December 15, 2015

14. Entergy letter to NRC, "Seismic Walkdown Report Revision 2 – Planned Update to Entergy's Response to NRC Request for Information Pursuant to 10 CFR 50.54(f) Regarding the Seismic Aspects of Recommendation 2.3 of the Near-Term Task Force Review of Insights from the Fukushima Dai-ichi Accident," (transmits Engineering Report WF3-CS-12-00003, Rev. 2), (ADAMS Accession No. ML14189A696), dated July 8, 2014.

15. NRC to All Licensees Holding Operating Licenses and Construction Permits for Nuclear Power Reactor Facilities, "Individual Plant Examination of External Events (IPEEE) for Severe Accident Vulnerabilities - 10 CFR 50.54(f)" Generic Letter No. 88-20, Supplement 4, (ADAMS Accession No. ML031150485), dated June 28, 1991

16. EPRI Technical Report TR-1016737, "Treatment of Parameter and Model Uncertainty for Probabilistic Risk Assessments," dated December 2008

17. NRC Regulatory Guide 1.147, "Inservice Inspection Code Case Acceptability, ASME Section XI, Division 1," Revision 19, (ADAMS Accession No. ML19128A244), dated October 2019

18. NRC Record of Review, "Dispositions to Waterford 3 Internal Events PRA Facts and Observations (F&Os)," (ADAMS Accession No. ML15363A374), dated October 6, 2015

19. NEI letter to NRC, "Final Revision of Appendix X to NEI 05-04/07-12/12-16, Close-Out of Facts and Observations (F&Os)," (ADAMS Accession No. ML17086A431), dated February 21, 2017

20. NRC Regulatory Guide 1.174, "An Approach for Using Probabilistic Risk Assessment in Risk-Informed Decisions on Plant-Specific Changes to the Licensing Basis," Revision 3, (ADAMS Accession No. ML17317A256), dated January 2018

21. NRC Report NUREG-1855, "Guidance on the Treatment of Uncertainties Associated with PRAs in Risk-Informed Decision Making," Revision 1, (ADAMS Accession No. ML17062A466), dated March 2017

22. EPRI Technical Report 1016737, "Treatment of Parameter and Modeling Uncertainty for Probabilistic Risk Assessments," dated December 19, 2008

23. EPRI Technical Update 1026511, "Practical Guidance on the use of Probabilistic Risk Assessment in Risk-Informed Applications with a Focus on the Treatment of Uncertainty," dated December 4, 2012

24. NRC Report NUREG-1921, "EPRI/NRC-RES Fire Human Reliability Analysis Guidelines," (ADAMS Accession No. ML12216A104), dated July 2012

25. NRC Report NUREG/CR-6850, EPRI 1011989, "EPRI/NRC-RES Fire PRA Methodology for Nuclear Power Facilities," (ADAMS Accession Nos. ML052580075, ML052580118), dated September 2005

26. Entergy Report, PSA-WF3-07-01, Waterford 3 Re-Examination of External Events Evaluation in the IPEEE, November 2017

27. Entergy Report, PSA-WF3-04-01, “Waterford 3 Seismic Risk Bounding Evaluation,” dated June 6, 2018

28. Entergy Report, PSA-WF3-08-06, Revision 0 – "PRA Technical Adequacy to Support Risk-Informed Applications,” dated November 11, 2020.

29. Entergy Report, PSA-WF3-08-02, "Waterford 3 Finding Level F&O Independent Technical Review,” dated December 11, 2018

30. Entergy Report, PSA-WF3-08-01, “Waterford 3 PRA Peer Review Gap Assessment to 2009 PRA Standard,” dated July 30, 2015

31. Entergy Report, PSA-WF3-01-QU, Revision 3 ,"WF3 PSA At-Power Level 1 Integration and Quantification Analysis," dated March 27, 2019

32. Entergy Report, PSA-WF3-01-QU-01, Revision 2 , "WF3 PSA Uncertainty and Sensitivity Analysis," dated February 27, 2019

33. Entergy Report, PSA-WF3-01-IF-SOU, "Waterford 3 Internal Flooding Sources of Uncertainty," dated May 6, 2020

34. Entergy Report, PSA-WF3-03-UNC-01, "WF3 Fire PRA Sensitivity and Uncertainty Report," dated December 23, 2019

35. Entergy Report, PSA-WF3-03-UNC-02, "Fire PRA Parametric Uncertainty Analysis (UnCert)," dated December 23, 2019

36. Entergy Report, PSA-WF3-03-FQ-01, "Waterford 3 Fire PRA Quantification Report," dated December 19, 2019

37. Entergy Report, PSA-WF3-03-ES-01, "Fire PRA Equipment and Cable Selection Notebook," dated December 19, 2019

38. Entergy Report, PSA-WF3-03-ES-02, "Fire PRA Circuit Analysis and Failure Probability Development," dated December 19, 2019

39. Entergy Report, PSA-WF3-03-PRM, "Fire PRA Quantification Model Preparation and Database Development," dated December 19, 2019

40. Entergy Report, PSA-WF3-03-FSS-02, "Waterford 3 PRA Fixed Ignition Source Zone of Influence Methods," dated December 19, 2019

41. Entergy Report, PSA-WF3-03-FSS-03, "Waterford 3 Transient Fire Scenario Report," dated December 19, 2019

42. Entergy Report, PSA-WF3-03-FSS-06, "Development of Fire Non-suppression Factors for WF3 Fire PRA Scenarios," dated December 23, 2019

43. Entergy Report, PRA-W3-01-IF-QU, "Waterford 3 Internal Flooding Quantification Report," dated May 6, 2020

44. Westinghouse Report, LTR-RAM-II-09-39, "RG 1.200 PRA Peer Review Against the ASME PRA Standard Requirements For The Waterford Steam Electric Station, Unit 3 Probabilistic Risk Assessment," dated August 17, 2009

45. Westinghouse Report, LTR-RAM-II-11-003 "Fire PRA Peer Review of Waterford Steam Electric Station Unit 3 Fire Probabilistic Risk Assessment Against the Fire PRA Standard Supporting Requirements from Section 4 of the ASME/ANS Standard," dated February 23, 2011

46. NRC letter to Entergy, "Waterford Steam Electric Station, Unit 3 - Issuance of Amendment Re: Adoption of TSTF-425, Revision 3 'Relocate Surveillance Frequencies to Licensee Control - RITSTF Initiative 5b' (CAC NO. MF6366)" (ADAMS Accession No. ML16159A419), dated July 26, 2016

47. NRC letter to Entergy, "Waterford Steam Electric Station, Unit 3 - Issuance of Amendment Regarding Transition to a Risk-Informed, Performance-Based Fire Protection Program in Accordance with 10 CFR 50.48(c) (CAC NO. ME7602)," (ADAMS Accession No. ML16126A033), dated June 27, 2016

48. Entergy Report, PSA-WF3-01-HR, Revision 3, "WF3 At-Power Human Reliability Analysis," dated March 27, 2019

49. Entergy Report, PSA-WF3-01-LE, Revision 3, "WF3 PSA Large Early Release Frequency (LERF) model," dated May 12, 2020

50. NRC Regulatory Issue Summary (RIS) 2007-06, "Regulating Guide 1.200 Implementation," dated March 22, 2007

51. NRC Letter to NEI, "U.S. Nuclear Regulatory Commission Acceptance on Nuclear Energy Institute Appendix X to Guidance 05-04, 07-12, AND 12-13, Close-Out of Facts and Observations (F&Os)," (ADAMS No. ML17079A427), dated May 3, 2017

52. Entergy Report, PSA-WF3-08-03, "Waterford 3 PRA Focused Scope Peer Review Review (IF & LERF)", dated March 23, 2020

53. Entergy Report, PSA-WF3-08-04, "Focused Scope Fire PRA Peer Review of Waterford Steam Electric Station Unit 3 Fire Probabilistic Risk Assessment Against the Fire PRA Standard Supporting Requirements from Section 4 of the ASME/ANS Standard," dated September 2012

54. NUREG/CR-6850 Supplement 1, "Fire Probabilistic Risk Assessment Methods Enhancements", dated September 2010

55. Entergy Report, PSA-WF3-08-05, "2nd Focused Scope Fire PRA Peer Review of Waterford Steam Electric Station Unit 3 Fire Probabilistic Risk Assessment Against the Fire PRA Standard Supporting Requirements from Section 4 of the ASME/ANS Standard," dated May 2013.

**Enclosure, Attachment 1**

**W3F1-2020-0047**

**List of Categorization Prerequisites**

## LIST OF CATEGORIZATION PREREQUISITES

Entergy will develop fleet level procedures to outline the process for categorization of plant systems. The Entergy fleet procedures will contain the elements/steps listed below for categorizing systems at Waterford 3.

- Integrated Decision-Making Panel (IDP) member qualification requirements.

- Qualitative assessment of system functions. System functions are qualitatively categorized as preliminary High Safety Significant (HSS) or Low Safety Significant (LSS) based on the seven criteria in Section 9 of NEI 00-04 (see Section 3.2 of the Enclosure). Any component supporting an HSS function is categorized as preliminary HSS. Components supporting an LSS function are categorized as preliminary LSS.

- Component safety significance assessment. Safety significance of active components is assessed through a combination of Probabilistic Risk Assessment (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. Safety-related components that are categorized as preliminary LSS are evaluated for their role in providing DID and safety margin and, if appropriate, upgraded to HSS.

- Review by the IDP. The categorization results are presented to the IDP for review and approval. The IDP reviews the categorization results and makes the final determination on the safety significance of system functions and components.

- Risk sensitivity study. For PRA-modeled components, an overall risk sensitivity study is used to confirm that the population of preliminary LSS components results in acceptably small increases to core damage frequency (CDF) and large early release frequency (LERF) and meets the acceptance guidelines of Regulatory Guide 1.174 (Reference 20).

- 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 of the Enclosure.

**Enclosure, Attachment 2**

**W3F1-2020-0047**

**Description of PRA Models Used in Categorization**

## DESCRIPTION OF PRA MODELS USED IN CATEGORIZATION

| Hazard | Model | Baseline CDF | Baseline LERF | Source |
|---|---|---|---|---|
| At Power Internal Events | Rev 6 | 3.03E-06 | 3.04E-08 | Reference 31 |
| Internal Fire | Rev 6 Fire | 2.01E-05 | 2.05E-07 | Reference 36 |
| Internal Flood | Rev 6 Flood | 1.49E-06 | 7.47E-09 | Reference 43 |
| **Total** | | **2.46E-05** | **2.43E-07** | |

**Enclosure, Attachment 3**

**W3F1-2020-0047**

# Disposition and Resolution of Open Peer Review Findings and Self-Assessment Open Items

# DISPOSITION AND RESOLUTION OF OPEN PEER REVIEW FINDINGS AND SELF-ASSESSMENT OPEN ITEMS

The Waterford 3 Internal Events PRA model (including internal flood) was peer reviewed in 2009 by the PWR Owners Group (PWROG). The review was conducted in May 2009 and the report was issued in August 2009 (Reference 44). The Internal Events PRA technical adequacy (including the 2009 peer review and follow up self-assessment results) has previously been reviewed by the NRC in previous LARs associated with TSTF-425 (surveillance frequency control program) and NFPA 805 license amendments (References 46 and 47).

The 2009 Peer Review was completed using the previous revisions of industry guidance. Revision 1 of Regulatory Guide 1.200 was applied, and the review was conducted using the 2005 issued PRA standard (Reference 9). In development of the NFPA 805 LAR, a gap assessment was performed due to revisions to the guidance and standard. The goal of this assessment was to both identify the differences in the two documents, and to identify the impact the differences potentially have on existing Waterford 3 peer review findings. The results of this gap assessment are documented in PSA-WF3-08-01 (Reference 30). Subsequent assessments were made relative to the 2009 PRA Standard (ASME/ANS RA-Sa-2009 – Reference 5) and Revision 2 of Regulatory Guide 1.200 (Reference 12).

The Fire PRA model was subject to a self-assessment and a full-scope peer review was completed in 2011. The review was in November 2010 and the resulting report was issued in February 2011 (Reference 45). Following a 2012 NRC Audit of the Waterford 3 NFPA 805 LAR and supporting documents, Waterford 3 revised the Fire PRA including several method changes. Waterford 3 completed two focused scope peer reviews (September 2012 and May 2013) to ensure proper evaluation of the revised methods.

The results of those peer reviews (for internal events and for the fire model) were the basis for the NFPA 805 LAR and SE and the TSTF-425 LAR and SE.

In the time following those reviews, the Waterford 3 PRA models have been through several updates (with some technical upgrades) as well as peer reviews and formal Facts and Observation Close-out reviews. Findings for both the full power Internal Events and Fire PRA models (as of October 2017) 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)," as accepted by NRC in the staff memorandum dated May 3, 2017 (Reference 51). The results of this review have been documented (Reference 29) and are available for NRC audit. The closure review concluded that the resolution of one finding resulted in application of a modeling upgrade. This one PRA upgrade as defined by the ASME PRA Standard RA-Sa-2009 (Reference 5) has occurred to the Internal Events PRA model since conduct of the PWROG peer review in 2009. That Upgrade is associated with cooling tower success criteria. It was reviewed during the F&O closure review in October 2017 (as an imbedded focused peer review).

Following the closure review, both the internal events model and fire PRA model were revised to address findings that were not closed during the closure review (in addition to a routine model update). During this update, the LERF model was revised with model upgrades to resolve issues and more thoroughly meet the PRA Standard. The LERF model was subject to a peer review in August 2019 as a result of the upgraded methods used. No unreviewed PRA

upgrades (except those noted for Flood and LERF) were included in the internal events and internal fire PRA revision efforts.

The Closure Review conducted excluded open findings associated with the internal flood PRA model. The model had not been thoroughly updated since the original findings. The Waterford 3 internal flood PRA model was revised in 2019. The flood PRA model update also included methodology upgrades. A peer review was necessary due to the method upgrades and the time gap from the previous flood model update (nearly ten years). The updated flood model was subject to a peer review in August 2019 for all PRA Standard elements relevant to internal flooding PRA models. Following the peer review, the flood PRA was updated in 2020 to address the peer review findings.

Table A3-1 lists the Peer Review efforts conducted in the past several years including dates and descriptions.

**Table A3-1 – Waterford 3 PRA Peer Reviews**

| Review Description | Review Document |
|---|---|
| RG 1.200 PRA Peer Review Against the ASME PRA Standard Requirements for the Waterford Steam Electric Station, Unit 3 Probabilistic Risk Assessment (Westinghouse Owner's Group August 2009) | LTR-RAM-II-09-39 (Reference 44) |
| Fire PRA Peer Review of Waterford Steam Electric Station Unit 3 Fire Probabilistic Risk Assessment Against the Fire PRA Standard Supporting Requirements from Section 4 of the ASME/ANS Standard (Westinghouse Owner's Group February 2011) | LTR-RAM-II-11-003 (Reference 45) |
| Focused Scope Fire PRA Peer Review of Waterford Steam Electric Station Unit 3 Fire Probabilistic Risk Assessment Against the Fire PRA Standard Supporting Requirements from Section 4 of the ASME/ANS Standard (Robert Brady - URS Corporation – September 2012) | PSA-WF3-08-04 (Reference 53) |
| 2nd Focused Scope Fire PRA Peer Review of Waterford Steam Electric Station Unit 3 Fire Probabilistic Risk Assessment Against the Fire PRA Standard Supporting Requirements from Section 4 of the ASME/ANS Standard (Robert Brady - URS Corporation – May 2013) | PSA-WF3-08-05 (Reference 55) |
| Waterford 3 Finding Level F&O Independent Technical Review (October 2017) | PSA-WF3-08-02 (Reference 29) |
| Waterford 3 Internal Flood/LERF Focused Scope Peer Review (August 2019) | PSA-WF3-08-03 (Reference 52) |

*Note – All of the Findings from the first four entries in the table were evaluated in the F&O Closure Review conducted in 2017. The only exception to this is the internal flooding and LERF related findings. All PRA Standard Elements for internal flooding and LERF were evaluated in the August 2019 flood/LERF peer review that was conducted on the updated flood model and LERF model.

Tables A3-2 through A3-4 provide a summary of the remaining findings and open items, including:

- Table A3-2 below lists the open Findings associated with the internal events PRA model and provides a disposition associated with the risk-informed applications for each open technical issue. This table includes open issues remaining from the 2017 closure review as well as the 2019 LERF related peer review.

- Table A3-3 below lists the open Findings associated with the August 2019 internal flood PRA model peer review.

- Table A3-4 below lists the open Findings associated with the internal Fire PRA model. This table addresses the findings that remain open following 2017 F&O Closure Review.

All three tables provide a disposition for each entry with regard to a risk-informed 10 CFR 50.69 program (and LAR).

The content of Tables A3-1, A3-2, A3-3, and A3-4 is contained in Waterford 3 PRA report PSA-WF3-08-06 (Reference 28). This includes listed references.

PRA Credit for FLEX

The Waterford 3 PRA model credits FLEX equipment and strategies. The credit for FLEX related equipment is limited to specific extended loss of offsite power scenarios and is limited to only permanently installed FLEX equipment. The model changes to incorporate FLEX equipment and strategies referenced site procedures and is a direct representation of the as-built, as-operated plant. The specific changes to add the FLEX equipment and strategies were not judged to be PRA upgrades, as existing modeling methods and techniques were used to update the model (no new or unique methods were applied).

The Waterford 3 PRA model credits a FLEX diesel generator to provide power to battery chargers (given an extended loss of offsite power). This diesel unit is installed in the Reactor Auxiliary Building (RAB). Use of this equipment and actions necessary to start and align it are included in site procedures for loss of offsite power, and all necessary equipment (cables, panels, keys, etc.) is pre-staged. Existing model failure data type codes were used for diesel generator and circuit breaker failure data. As the equipment is permanently installed and procedurally controlled, generic failure data was judged applicable.

The FLEX Core Cooling Pump (FCCP) is a permanently installed pump that can be aligned to either provide Feedwater to the Steam Generators (SGs), makeup to the RCS, or backup cooling to the Spent Fuel Pool. The pump can be powered from a charging pump breaker, supplied by emergency power, or powered from the FLEX Diesel. For the Waterford 3 PRA model, only alternate Feedwater to the Steam Generators is credited. As with the FLEX diesel, all equipment and actions necessary to align and operate the pump for this function are driven by site procedures. Existing model failure data type codes were used for the FCCP. As the equipment is permanently installed and procedurally controlled, generic failure data was judged applicable.

The human actions added to the PRA model for FLEX deployment followed the same HEP development methods as all other modeled actions. Credited actions are all procedure driven actions. Peer reviewed HEP methodology was applied to the added actions. Live timed field trials were used to support timing inputs for HEP development. The credited operator actions are procedure driven actions and are similar to other operator actions evaluated using

approaches consistent with the ASME/ANS PRA Standard as endorsed by RG 1.200 Revision 2.

The modeling of the credited FLEX equipment and actions:

- Does not represent any new methods;
- Does not change the scope of the model given that the equipment, dependencies, and type of accident sequences remain the same;
- Does not represent a change in capability of the PRA model given the original and updated models can both evaluate the risk associated with loss-of-offsite power and station blackout.

The changes implemented for the incorporation of the FLEX modifications were within the framework of the existing peer reviewed PRA model structure.

| Table A3-2 **– – Open Internal Events Peer Review Findings Assessed During F&O Closure Review** | | | | |
|---|---|---|---|---|
| **Finding Number** | **Supporting Requirement(s)** | **Capability Category (CC)** | **Description** | **Disposition for 10 CFR 50.69** |
| **AS-B3-01** | AS-B3 | Not Met | The AS report (PRA-W3-01-001S01 Revision 1) includes discussion of the phenomenological impacts of heating of the containment sump water (failure of HPSI recirculation due to loss of required NPSH and pump cavitation) and large containment rupture (loss of safety injection due to the rapid depressurization, flashing of hot water in the sump, and loss of net positive suction head to the HPSI pumps) that can occur due to inadequate containment heat removal. However, some events such as steamline breaks and feedwater line breaks can result in harsh environments (especially steam and high temperature) where mitigating equipment are located.<br><br>Phenomenological impacts must be considered in order to ensure risk results are not underestimated.<br><br>Consider and document the phenomenological conditions from the entire range of initiating events, especially high energy line break. | This was graded by the 2017 F&O closure review team as partially resolved (with open documentation issues).<br><br>WCAP-16679-P – 'Accident Sequence Phenomena' was reviewed to determine if any additional phenomena needed to be addressed in the current Waterford 3 AS analysis. All other phenomena have been addressed in the accident sequence and the system analyses, as necessary. The effects of steam line and feed line breaks are evaluated in the initiating event document.<br><br>Waterford 3 completed a review of the phenomenological considerations in the AS report immediately following the original peer review (in 2010). That analysis and the results of it were not included in (or referenced) in the model Revision 5 AS report reviewed during the closure review. The considerations are included in the model – the updated documentation was noted to be insufficient. This disconnect in documentation was the basis for grading this finding as partially resolved with open documentation issues.<br><br>The Revision 6 update has a more thorough documentation of such phenomena and their treatment. The Revision 6 Accident Sequence and Success Criteria documentation contain the necessary details to satisfy AS-B3 (though a formal closure review has not been completed).<br><br>This Finding has no impact on quantified results and no impact on implementation of a 10 CFR 50.69 program. |
| **HR-B1-01** | HR-B1 | CC-I | There is no pre-initiator identified for CCW, because of the CCW is a running system. However, the CCW system may support the safety related standby system. The path of the CCW to support this system may be failed due to pre-initiator HFE. The potential for this type of pre-initiator HFE needs to be re-evaluated and documented. | This was graded by the 2017 F&O closure review team as partially resolved (with open documentation issues).<br><br>This F&O has been resolved during the 2019 Revision 6 model update with a thorough review of pre-initiators and evaluation of the exact consequence of each failure. Restoration errors of CCW to a standby system are included in the restoration of the associated standby system. For instance, CCW to the Containment Spray pumps is included in the Containment Spray restoration logic (not the CCW logic). Therefore, these restoration errors have been identified and evaluated in the current model (and correctly modeled to the right system consequence). |

| Table A3-2 – – Open Internal Events Peer Review Findings Assessed During F&O Closure Review | | | | |
|---|---|---|---|---|
| **Finding Number** | **Supporting Requirement(s)** | **Capability Category (CC)** | **Description** | **Disposition for 10 CFR 50.69** |
| | | | | The closure review team judged that support system related pre-initiators are properly accounted for and modeled but the system notebooks were not explicit enough in the detailed explanation.<br><br>The Revision 6 PRA model documentation contains the necessary details and more thoroughly documents pre-initiator modeling. This has no impact on the 10 CFR 50.69 program. This is a documentation issue only and will have no impact on quantified results. |
| **HR-F2-01** | HR-F2 | CC-I | The cue of each HFE is not clearly addressed. The timing of each cue should be included in the description of the HFE time windows and included in the quantification of the HEPs. | This was graded by the 2017 closure review team as partially resolved.<br><br>For HEP events, the cues (i.e., annunciators, EOP/AOP entry conditions) are explicitly discussed in each operator action in the model and are documented in the operator interview sheets. All of this is documented in the updated PRA HRA analysis PSA-WF3-01-HR (Reference 48).<br><br>The closure review team assessed that the HEP development spreadsheets (which are used or referenced in the HR analysis) lacked thorough detail and references associated with operator timing and cues. The timings used were/are reasonable and references/operator interviews could be found, but a direct tie in the development documentation was not always explicitly provided. The spreadsheets used to calculate the HEP values lacked basis for all timings used in development. This limitation resulted in the finding remaining open.<br><br>The PRA Revision 6 update included application of the EPRI HRA calculator. The information associated with this finding is now more thoroughly documented for each modeled action.<br><br>This has no impact on the 10 CFR 50.69 program. This is a documentation issue only and will have no impact on quantified results. |
| **HR-G4-01** | HR-G4 | Not Met | As seen in the HRA spreadsheet (hfe_cp.xls), the time available to complete actions is based on a range of references including plant-specific calculations. However, in some cases unjustified and/or inaccurate assumptions were used as a basis. The event timelines in the HRA spreadsheets also do not consistently identify the specific point in time relevant indications are received. | This was graded by the 2017 closure review team as partially resolved (with open documentation).<br><br>For HEP events, the cues (i.e., annunciators, EOP/AOP entry conditions) are explicitly discussed in each operator action in the model and are documented in the operator interview sheets. All of this is documented in the updated PRA HRA analysis PSA-WF3-01-HR (Reference 48). |

| Table A3-2 – – Open Internal Events Peer Review Findings Assessed During F&O Closure Review | | | | |
|---|---|---|---|---|
| **Finding Number** | **Supporting Requirement(s)** | **Capability Category (CC)** | **Description** | **Disposition for 10 CFR 50.69** |
| | | | Inaccurate HFE time windows may result in under- or over-estimation of HEPs. Ensure the time windows are justified based on realistic analyses or simulation and identify in the HEP worksheets the point in time relevant indications are received. | The closure review team assessed that the HEP development spreadsheets (which are used/referenced in the HR analysis) lacked thorough detail and references associated with operator timing and cues (and timing of specific cues). The timings used were/are reasonable and references/operator interviews could be found, but a direct tie in the development documentation was not always explicitly provided.<br><br>The 2019 PRA Revision 6 update included application of the EPRI HRA calculator. The information associated with this finding is now more thoroughly documented for each modeled action.<br><br>This has no impact on the 10 CFR 50.69 program. This is a documentation issue only and will have no impact on quantified results. |
| **SC-C1-02** | SC-C1 | Not Met | Throughout the document there are a number of assumptions and statements made that directly impact the success criteria but do not have any references identified to justify their bases. Querying the PRA group determined that most of the statements were based on valid references, but they were not identified in the success criteria documentation. The references need to be specifically identified and included. | This was graded by the 2017 closure review team as partially resolved.<br><br>The review team concluded that, while the revised report did contain some improvement in providing references/bases for assumptions and SC treatments, it is still not sufficient to meet CC-II of the Standard. This includes some missing references, duplicate entries, and references to old/superseded documents.<br><br>The PRA model Revision 6 update included an updated SC report with a detailed review of references to ensure a valid basis was provided for documented success criteria.<br><br>This has no impact on the 10 CFR 50.69 program. This is a documentation issue only and will have no impact on quantified results. |
| **SY-A12b-01** | SY-A12 (SY-13)<br><br>SY-B14 (SY-15) | Not Met | Need to use the exclusion criteria in SY-A14 to justify excluding flow diversion pathways. Using the criteria 2 normally closed valves should be easily justified using criteria SY-A14(a). The criteria for excluding based on a 1 to 3 ratio between the primary piping and the potential diversion piping needs to be backed up by pressure differentials. This exclusion criteria is valid if the system pressures between the primary and potential diversion piping is the same or similar. If the pressure differential is high, further analysis is required to justify exclusion. | This was graded by the 2017 closure review team as partially resolved.<br><br>The 1/3 exclusion criteria is noted as standard treatment for systems, unless otherwise noted. However, it is not uniformly applied to all systems. Systems with different modeling treatment and the basis for the different treatment are noted in the documentation. Additional flow diversion failures were credited following this finding for HPSI and LPSI based on meeting 1/3 criteria but having high pressure differential, and CCW Makeup for having a finite and limited volume. |

| Table A3-2 – – Open Internal Events Peer Review Findings Assessed During F&O Closure Review | | | | |
|---|---|---|---|---|
| **Finding Number** | **Supporting Requirement(s)** | **Capability Category (CC)** | **Description** | **Disposition for 10 CFR 50.69** |
| | | | Overall, the assumptions used to exclude specific types of failures needs to be reevaluated and justification provided on how the exclusion criteria is met. | The CCW diversion would not fail system function but over time would reduce inventory below functional level before 24 hours.<br><br>This element has been resolved in the Revision 6 PRA model documentation but closure through a formal review is not yet complete.<br><br>This has no impact on the application of a 10 CFR 50.69 program. |
| **LE2-3**[(1)] | LE-E1*<br><br>LE-C9<br><br>*Finding Reference this SR but it is noted in the Report as 'Met' | Met<br><br>Not Met | The LERF model includes events reflecting conditional probability of the Component Cooling System (CCS) and the Containment Spray (CSS) due to a harsh severe-accident environment. Section 8.1.9 of the LERF Notebook (PSA-WF3-01-LE, Rev. 2) identifies events P_CCSFAILS and P_CSSFAILS and refers to the Level 2 analysis for the development of the associated conditional probabilities. The Level 2 report (PSA-WF3-01-L2-01, Rev. 0), in turn, refers to the Waterford 3 Individual Plant Examination (IPE, Waterford 3 Probabilistic Risk Assessment Individual Plant Examination Submittal, August 1992) as the source. Table 4.6-2 of the IPE lists values for these events but provides no further analysis or justification.<br><br>The SR requires that justification be provided for continued operation of systems under adverse (severe-accident) environments. Although the potential for an elevated probability of failure has been incorporated for the CCS and CSS, no justification for the probabilities selected is provided.<br><br>Develop a justification for the continued operation of CSS and CCS under adverse conditions. | Continued operation of CCS and CSS under harsh conditions/environment has an insignificant impact on overall LERF results.<br><br>Vessel Rupture accounts for over 90% of total LERF. ISLOCA is the second largest contributor accounting for over 50% of the non-Vessel Rupture LERF total. CCS and CSS have no impact on these two LERF contributors. The values applied to events P_CCSFAILS and P_CSSFAILS do not significantly impact LERF results. The revised LERF report shows a sensitivity case with the applied CSS and CCS values increased by a factor of 10. This change results in a less than 1% increase in LERF.<br><br>Solving the fault tree for the 'loss of train A and B containment spray system' gate generates a 1.7E-03 result. The applied P_CSSFAIL is 5E-03. The LERF input for loss of CSS (originally from the IPEE) is nearly a factor of 3 higher than the detailed fault tree logic. Similarly, the CCS value is nearly a factor of 10 higher than calculated in the fault tree.<br><br>The systems in question are both designed to operate in a harsh environment, and most of the active components for the system are outside containment (and won't be impacted by a harsh containment environment).<br><br>The model revision 6 LERF report (PSA-WF3-01-LE, Reference 49) has been updated to provide additional basis for the values used. The applied values are conservative and have no significant impact on LERF results. This finding has no impact on quantified results and will not impact risk informed applications. |
| **LE2-4**[(1)] | LE-E2<br><br>LE-F2* | Not Met<br><br>Cat-I/II/III | The treatment of accident-progression phenomena relies almost entirely on predictions from MAAP calculations. While this is appropriate for much of the LERF analysis, the reliance on MAAP for crediting that core debris remains in the reactor pressure vessel for a preponderance of | The LERF analysis has been updated to include additional discussion relevant to the limitations of MAAP 4.0.6, as well as to include additional sensitivities regarding in-vessel melt retention issues (and several specific sensitivity cases related to the MAAP 4.0.6 limitation associated with this phenomena). The updated documentation |

| Table A3-2 – – Open Internal Events Peer Review Findings Assessed During F&O Closure Review | | | | |
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| **Finding Number** | **Supporting Requirement(s)** | **Capability Category (CC)** | **Description** | **Disposition for 10 CFR 50.69** |
| | *Finding References this SR but it is noted in the Report as 'Met' | | scenarios is a potentially significant non-conservatism. It is difficult to determine what the impact on LERF would be if these scenarios were permitted to lead to relatively early breaching of the vessel.<br>The reasonableness check did not address the appropriateness of MAAP 4.0.6 for certain accident phenomena, including in-vessel melt retention.<br><br>Investigate the potential for in-vessel retention in more detail, based on available literature and/or a tool that is suitable for assessing this potential. Adjust the LERF analyses to reflect more realistic treatment of in-vessel melt retention.<br><br>Ensure that the treatment of in-vessel melt retention is considered during future checks of the reasonableness of LERF contributors. | provides more detail on in-vessel retention and addresses the uncertainty of the cases and computer codes used in the evaluation.<br><br>The following is an excerpt from the revised LERF document:<br>*"Relative to the ability of MAAP 4.06 to model in-vessel retention, it is important to be aware of a Trouble Report posted for MAAP 4.08 and earlier releases. The error report indicates the following:*<br><br>*Reference: TR 810 Ex-vessel cooling code error (Mod. Package 833)*<br>*As indicated in the reference above, there is a code error in MAAP4.06 related to ex-vessel cooling:*<br><br>*If the reactor cavity is flooded, there is substantial debris in the lower head, and ex-vessel cooling is enabled, it is possible to melt through the reactor wall heat sink (i.e. mrvn(6,1) through mrvn(6,5) become 0), and the reactor vessel has not failed. If this occurs, the tabular output file will show primary system energy imbalance, and RV failure will not have occurred for the MAAP sequence.*<br><br>*To address this error report, a sample of the existing MAAP cases indicating that vessel failure was prevented were reviewed to better understand if this error is occurring for those runs. As stated in the TR description above, the two vessel wall heat sinks should be examined to determine if the vessel had indeed failed, but that indication of breach was not acknowledged. In order to make this determination, those cases may need to be rerun to include plot variables associated with the heat sink mass as identified in the TR. The Modification package includes a work-around that can be added to the input file to correct this error. For the selected sample cases, the mass of the heat sinks did not become zero as discussed above and therefore the error was not a factor in the determination of vessel breach."*<br><br>The LERF documentation has been updated to more thoroughly address the issues identified in the Finding. This finding does not affect the base model, has no impact on quantified results, and will not impact risk informed applications. |
| **LE4-4**[(1)] | LE-G5 | Not Met | Section 9.6 of PSA-WF3-01-LE, Rev. 2 lists the types of PRA limitations to consider from the ASME/ANS PRA Standard but provides no specific discussion of the actual limitations of the Waterford 3 LERF model nor how they | The LERF documentation has been updated to address this Finding and detail the impact of LERF assumptions on PRA applications. Assumptions related to model development can and do impact results. However, the overall methods and process used limit those |

| Table A3-2 – – Open Internal Events Peer Review Findings Assessed During F&O Closure Review | | | | |
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| **Finding Number** | **Supporting Requirement(s)** | **Capability Category (CC)** | **Description** | **Disposition for 10 CFR 50.69** |
| | | | may impact risk-informed applications. Attachment A identifies the assumptions in the Waterford 3 LERF model, but does not identify impacts on applications from the treatment of the assumptions.<br>Document the model limitations and potential impacts on PRA applications for Waterford 3 LERF model. | impacts. Consensus modeling approaches are used in PRA model development. The LERF model is also peer reviewed against the PRA Standard. Sensitivity cases for key sources of uncertainty have been developed to ensure the impact of assumptions and modeling decisions are known and documented. This thorough, peer reviewed, state-of-the-art approach to LERF modeling helps ensure the model and results maintain the technical adequacy requirements to support risk-informed applications.<br><br>The LERF documentation has been updated to more thoroughly address the issues identified in the Finding. This finding has no impact on quantified results and will not impact risk informed applications. |
| **LE4-5**[(1)] | LE-F2*<br><br>LE-D6*<br><br>*Finding References this SR but it is noted in the Report as 'Met' | Met Cat I/II/III<br><br>Met Cat III | An analysis using the MAAP 4.0.6 computer code based on realistic inputs in PSA-WF3-01-LE, Rev. 2, calculates the plant specific conditions for the TISGTR accident sequences. The TISGTR analysis relies solely on MAAP 4.0.6 analyses to make the determination of whether an accident progression sequence is classified as TISGTR and does not apply split fractions based on industry guidance. The capabilities of the MAAP 4.0.6 code should be validated for the appropriate use for this determination. Plant procedures are incorporated into the operator actions governing the TISGTR response.<br><br>The reasonableness check did not address the appropriateness of MAAP 4.0.6 for certain accident phenomena, including TISGTR. | The LERF documentation has been updated to more thoroughly address the appropriateness of MAAP 4.0.6 as it applies to TISGTR.<br><br>The updated reasonableness check is based on:<br>*EPRI Perspective on Thermally-Induced Steam Generator Tube Rupture Issues, NRC document ID: ML071340053, Marc Kenton.*<br><br>It is commonly believed that the TISGTR has a minor impact on the PRA results. Hot leg creep rupture has been found to either occur prior to tube rupture or immediately after. In both instances, the dominant flow path will be from the reactor vessel to containment, with a relatively small flow through the failed SG tube.<br><br>Based on the reference above, the MAAP code can accurately predict the peak temperature in the SG tubes as well as the hot leg.<br><br>The LERF report was revised to update the reasonableness argument (including addition of the noted reference) to address this Finding. This Finding has no impact on quantifies LERF results or application of the model for the 10 CFR 50.69 program.<br><br>Note – The two SRs listed in the Finding are both noted in the report as 'met' for Capability Category II or III. Even though the finding was issued the grading of the fining is that the analysis is sufficient for PRA applications as assessed (with a recommendation for improvement – which was completed). |
| Note 1 – These findings are the result of a peer review of the LERF model conducted in August 2019. The other listed findings are the ones that remained open following an F&O Closure Review in 2017. | | | | |

| Table A3-3 – **Internal Flooding Open Peer Review Findings** | | | | |
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| **Finding Number** | **Supporting Requirement(s)** | **Capability Category (CC)** | **Description** | **Disposition for 10 CFR 50.69** |
| **FL1-3** | IFSN-A15 | Met CC-I/II/III | Table B.3 of PSA-WF3-01-IF-WD lists piping segments (flood sources) that require further analysis as to whether they could be screened. A Waterford 3 PRA Model Change Request (MCR #W3-6480) was written to track this issue.<br><br>Complete the analysis of these piping segments to determine whether they can be screened per the criteria given in this SR. | The pipe segments excluded from the analysis have been identified and evaluated. The updated Flood PRA documents account for the subject segments. Many screened but several resulted in new or updated scenarios.<br><br>The updated flood analysis properly treats the segments identified in this Finding. This issue has no impact on quantified results and will not impact risk-informed applications.<br><br>Though a Finding was issued, the peer review graded the SR as Met CC-I/II/III. |
| **FL-1-4** | IFQU-B3*<br><br>*Finding References this SR but it is noted in the Report as 'Met' | Met | In Table A-2 of PSA-WF3-01-IF-SOU, item IF-C-9 dealing with the structural analysis of doors may have a significant impact on internal flood CDF. The configuration of doors within room 211 (vestibule area) may be such that the door leading to the outside environs may preferentially fail first, which could have a significant impact on the calculation of internal flood CDF (~50%). | A detailed engineering evaluation of the subject door was completed by Waterford 3/Entergy structural engineers. The Flood PRA scenarios were updated based on this more thorough evaluation of flood propagation from room 211. The Flood PRA was updated to explicitly address this Finding. Resolution of this issue did result in a significant change in flood results. With the issue resolved, this Finding has no further impact on quantified results and will not impact risk-informed applications.<br><br>This Finding was issued even though IFQU-B3 was graded as "Met Cat 1-3" in the peer review report. |
| **FL 1-6** | IFEV-A5*<br><br>*Finding References this SR but it is noted in the Report as 'Met' | Met CC-I/II/III | The frequency for event %FLD-TB_ALL used an older data set (NUREG/CR-5750) that was inconsistent with the data used throughout the rest of the plant. The EPRI reference that was primarily used for other event frequencies outside the Turbine Building can also be used to develop plant-level flood frequencies based on generic industry operating experience.<br><br>This SR refers to the requirements in Section 2-2.1 of the Standard, which does involve SR IE-C12 that requires a comparison of results of the initiating event analysis with generic data sources to provide a reasonableness check of the results. The event frequency used from NUREG/CR-5750 is an older data source that was not compared with the more recent data set used for the rest of the plant.<br><br>Compare the event frequency used for the Turbine Building floods with the plant-level data found in Section 8 of EPRI 3002012997, which covers a larger time period | The Flood documentation has been updated. Updated initiating event frequencies have been developed (in part to address Finding IFEV-A6 listed below in this table). Additional details on the development of the Turbine Building frequency (%FLD-TB_ALL) have been added. Reasonableness checks of resulting frequencies were completed to ensure compliance with Standard SR IE-C12 (and IFEV-A5).<br><br>This Finding has been addressed in the updated Flood PRA documentation. This issue has no impact on quantified results and will not impact risk-informed applications.<br><br>This Finding was issued even though IFEV-A5 was graded as "Met Cat 1-3" in the peer review report. |

| Table A3-3 – **Internal Flooding Open Peer Review Findings** | | | | |
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| **Finding Number** | **Supporting Requirement(s)** | **Capability Category (CC)** | **Description** | **Disposition for 10 CFR 50.69** |
| | | | than the period used in NUREG/CR-5750. This will confirm the "reasonableness" of the selected event frequency for this initiating event and whether it should be deemed conservative as stated in Section 2.2.146. | |
| **FL3-2** | IFPP-B1*<br><br>*Finding References this SR but it is noted in the Report as 'Met' | Met | There are discrepancies between what was modeled in PSA-WF3-01-IF-AS and the information in Table 2 of PSA-WF3-01-IF-FA. For example, RAB-21-211 lists Door D11 to RAB21-Q as not having sufficient accumulation of water to fail, and PRA equipment damage protected by D11 is listed as 'N/A'.<br><br>Tables 1 and 2 of PSA-WF3-01-IF-FA should be updated to be consistent with the accident sequence analysis described in Section 4 of PSA-WF3-01-IF-AS. | The Flood PRA documentation has been updated to resolve discrepancies between the Accident Sequence and Flood Area Development documents. The revised documents and tables contain consistent technical information.<br><br>The Flood PRA was updated to explicitly address this Finding. With the issue resolved, this Finding has no impact on quantified results and will not impact risk-informed applications.<br><br>This Finding was issued even though IFPP-B1 was graded as "Met Cat 1-3" in the peer review report. |
| **FL3-4** | IFEV-A6 | Cat I | EPRI TR-3002012997 R4 Section 4.5 requires consideration of age correction for significant scenarios. Aging factors from Table 4-19 were not applied. It could be argued that age adjustment using EPRI TR-3002000079 R3 would not be applicable to Waterford 3 for another 2 to 3 years.<br><br>This SR requires consideration of material condition of fluid systems. EPRI TR-3002000079 R3 does not require age correction for FP piping that has been in service less than 40 years. However, EPRI TR-3002012997 R4 Table 4-19 provides age correction factors ranging from 10 to 50 years of service. Frequencies calculated in PRA-WF3-IF-QU were based on nominal values from EPRI TR-3002012997 R4 without age correction. | EPRI has rescinded the TR-3002012997 R4 guidance document. The flood PRA initiating events were updated to use EPRI TR-3002000079 R3 (which is still endorsed by EPRI). Following the TR-3002000079 R3 guidance does not require age correction due to the age of the Waterford 3 site and piping systems.<br><br>The Flood PRA was updated to explicitly address this Finding. With the issue resolved, this Finding has no impact on quantified results and will not impact risk-informed applications. |
| **FL3-5** | IFEV-A7 | Not Met | Section 2.1.1 of PRA-WF3-01-IF-IE includes an invalid input to the equation used to calculate the frequency of a maintenance induced flood event. The use of a maintenance unavailability value (or probability) was erroneously used as a frequency in the justification for screening maintenance induced flooding events from the analysis.<br><br>This SR requires consideration of human-induced floods from maintenance. While consideration of maintenance induced flooding, events was provided in Section 2.1.1 of | The flood documentation has been updated to correct the issue identified (including correction of the erroneous value) in this Finding. Human induced flood events are considered in the Waterford 3 flood PRA analysis. Human induced flood events have been screened, and a valid basis for the screening is included in the updated documentation.<br><br>The Flood PRA was updated to explicitly address this Finding. With the issue resolved, this Finding has no impact on quantified results and will not impact risk-informed applications. |

Table A3-3 **– Internal Flooding Open Peer Review Findings**

| Finding Number | Supporting Requirement(s) | Capability Category (CC) | Description | Disposition for 10 CFR 50.69 |
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| | | | PRA-WF3-01-IF-IE, the basis for screening these events is invalid.<br><br>Correct the qualitative argument to justify screening human induced flood events based on a review of generic industry data or apply the methodology given in Section 8 of EPRI 3002012997 R4. | |
| **FL3-7** | IFQU-A10 | Not Met | Overlaying the internal flood logic and target sets onto the LERF sequences produced erroneous cutsets relative to the corresponding CDF cutsets. For example, the top LERF cutset from flooding scenario %FLD-RAB_21-211FP-L includes AB Switchgear alignment events 'B_TO_AB' and 'HPIABISSTBY' while the corresponding CDF cutset is more consistent with loss of equipment in the AB and A Switchgear Rooms.<br><br>Justify the validity of the anomalous LERF cutsets generated by internal flood initiators or modify the LERF analysis as necessary to account for the unique flood-induced scenarios in accordance with the applicable requirements described in 2-2.8. | The Flood PRA was updated following the peer review. The update included changes to key scenarios and initiating event frequencies. The update required updated quantification of CDF and LERF. During the update, an existing modeling error (not explicitly related to flood) was found and corrected. CDF and LERF cutsets were reviewed for reasonableness and to ensure mapping/overlaying the CDF results onto LERF was properly completed.<br><br>Updated Flood LERF results were reviewed to check for erroneous results. With the issue resolved, this Finding has no impact on quantified results and will not impact risk-informed applications. |

Note – All Flood PRA findings are the result of an August 2019 Flood PRA peer review.

| Table A3-4 – **Fire PRA Open Peer Review Findings Assessed During F&O Closure Review)** | | | | |
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| **Finding Number** | **Supporting Requirement(s)** | **Capability Category (CC)** | **Description** | **Disposition for 10 CFR 50.69** |
| **CS-A3-01** | CS-A3<br><br>ES-B4 | Met<br><br>Not Met | The component and cable selection report, Rev. 0, reviewed looking for effects of interlocks and permissive. There is discussion of the VCT in the MSO discussion, but there does not appear to be any other interlock or permissive discussion. These items would be tank level interlocks with valves and pumps, loss of one CCW, perhaps a flow switch that starts the pump in the other train. Various interlocks that are associated with the starting of a pump such as adequate lube oil pressure, cooling water flow, etc.<br><br>The instruments and cables associated with permissives and interlocks do not appear to have been comprehensively addressed in the PRA. Starting interlocks for pumps and breaker closure or tank interlocks that open or close valves or flow switches that start pumps all could have fire effects that would adversely affect the success of various system functions. | The 2017 F&O Closure review judged this finding as 'Open'.<br><br>The Fire PRA model and documentation was updated in 2019 to incorporate the internal event (R6) model update and to address open Findings.<br><br>The mapping of all items on the SSEL was reexamined with attention paid to instruments to ensure that their consequential impacts have been properly linked to the PRA model. The methodology and results of the analysis are documented in the Waterford 3 Component and Cable Selection Report (PRA-W3-03-ES-01, Reference 37).<br><br>Specific rationale of impacts on permissives and interlocks is documented for several components and cables in the MSO Expert Panel discussions (PRA-W3-03-ES-01, Reference 37).<br><br>The updated documentation references the site Nuclear Safety Capability Assessment (NSCA). The NSCA contains a thorough, comprehensive review and treatment of interlocks and permissives. This finding has been addressed in the 2019 model update (though it has not been through formal closure review). This finding has no impact on PRA results or risk informed applications. |
| **CS-B1-01** | CS-B1 | CC-I | Electrical coordination is addressed in the scenario development report (R0247070001.06 Appendix E). Appendix E of R0247070001.06 provides information concerning electrical coordination. However, it is incomplete because the supplemental coordination evaluation is missing from the document. Preliminary coordination review has been performed and exists in an email (though not formally documented).<br><br>Appendix E of R0247070001.06 provides information concerning electrical coordination. However, it is incomplete because the supplemental coordination evaluation is missing. Information was provided by email and needs to be formally incorporated, (finding CS-B1-01) CC II/III when finished. | The original finding was judged by the F&O closure teams as Open.<br><br>Breaker coordination is addressed in the updated Fire PRA model documentation. An updated analysis that supplements the Component and Cable Selection report and the Plant Response Model report has been developed. The analysis (PSA-WF3-03-ES-02, Reference 38) provides a more thorough analysis of electrical coordination.<br><br>The updated model and documentation provide a thorough resolution to the breaker coordination issue. This finding has been addressed in the 2019 model update. This finding has no impact on PRA results or risk informed applications. |
| **ES-A2-01** | ES-A2 | Not Met | R0247070001.02, Rev. 0, reviewed. Three issues identified: 1. The loss of equipment due to a loss of room cooling caused by a fire damper going closed could not be demonstrated. 2. Although the R0247070001.06 credited | The original finding was judged by the 2017 F&O closure teams as 'Open'. |

| Table A3-4 – **Fire PRA Open Peer Review Findings Assessed During F&O Closure Review)** | | | | |
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| **Finding Number** | **Supporting Requirement(s)** | **Capability Category (CC)** | **Description** | **Disposition for 10 CFR 50.69** |
| | | | breaker coordination from the SSD analysis, the PRA model does not appear to address the effect of a fire interrupting the relay circuit that would inhibit the coordination and allow a fault to transfer upstream.<br>3. The loss of DC from a fire does not appear to be addressed fully. For example, a fire induced loss of DC to the RCP breakers would inhibit the operator action to trip the RCPs from the control room. Note: Items 1 and 3 have outstanding questions to Waterford 3. | The first item identified in the Finding is no longer an issue. Room cooling has been removed from the model based on room heatup evaluations that determined cooling was not needed to support operation through the mission time. Room cooling is no longer in the PRA model except for the Main Control Room.<br><br>DC breaker impacts to the RCPs have been explicitly addressed and a (ex-control room) Recovery Action was added to the model to address cases where fire damage could impact the ability of the operators to trip the pumps from the control room.<br><br>Breaker coordination (including DC breakers) is addressed in an analysis that supplements the Component and Cable Selection report and the Plant Response Model report. The analysis (PSA-W3-03-ES-02, Reference 38) provides a more thorough analysis of electrical coordination and describes modeling that was added to the PRA to correctly account for DC power requirements for overcurrent protection.<br><br>The updated model and documentation provide a thorough resolution to the breaker coordination issue. This finding has been addressed in the 2019 model update. This finding has no impact on PRA results or risk informed applications. |
| **ES-A3-02** | ES-A3<br><br>CS-A3 | CC-I/II/III<br><br>Met | The loss of DC does not appear to be adequately addressed in the fire PRA. For example, a failure of DC to supply control power to the RCP breakers would inhibit the operator action to trip the Reactor Coolant Pumps (RCPs) in a loss of seal cooling scenario. This was compensated for by a spurious start of the RCPs which would affect the same state in the model. Similarly, a loss of DC power could potentially transfer a fault due to inhibition of coordination.<br><br>The plant has redundant DC supplies to the two breakers which makes this failure less probable. However, additional documentation is required to clarify the issue. The fire effects on DC could adversely affect coordination as well as remote operation of breakers. | The original finding was judged by the 2017 F&O closure teams as partially resolved.<br><br>DC breaker impacts to the RCPs have been explicitly addressed and a (ex-control room) Recovery Action was added to the model to address cases where fire damage could impact the ability of the operators to trip the pumps from the control room.<br><br>Breaker coordination (including DC breakers) is addressed in an analysis that supplements the Component and Cable Selection report and the Plant Response Model report. The analysis (PRA-W3-03-ES-02, Reference 38) provides a more thorough analysis of electrical coordination and describes modeling that was added to the PRA to correctly account for DC power requirements for overcurrent protection.<br><br>The updated model and documentation provide a thorough resolution to the breaker coordination issue. This finding has been addressed in the 2019 model update. This finding has no impact on PRA results or risk informed applications. |

| Table A3-4 – **Fire PRA Open Peer Review Findings Assessed During F&O Closure Review)** | | | | |
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| **Finding Number** | **Supporting Requirement(s)** | **Capability Category (CC)** | **Description** | **Disposition for 10 CFR 50.69** |
| **ES-C2-01** | ES-C2 | Not Met | Component and Cable Selection Report R0247070001.02, Revision 0 in Section 2.6 states, "An instrumentation review was conducted using the simulator and operators to identify single instrument reliance and single indication/instruments whose malfunction would cause operators to take action that would result in un-recoverable states. No single instrument vulnerabilities were identified."<br><br>However, there is no documentation or discussion of this activity. Engineering standard EN-FP-S-008-Multi has a process for reviewing indication needs for post fire in the simulator in 5.3.4, but this does not specifically address spurious indications that cause unwanted actions.<br><br>What is needed to meet Category II for this SR is to develop a process for how various indications are reviewed and screened and then considered for inclusion into the FPRA model. There is a sample process that the PWROG did for ERGs for Westinghouse sites, this process is more detailed than required for meeting this SR for this application but does show the process. No evidence other than statement that a simulator walkdown was performed. | The original finding was judged by the 2017 F&O closure teams as 'partially resolved.<br><br>The Fire PRA specific operator interviews (Engineering Change EC 46718) have the following statement: With respect to the potential for undesired operation actions or errors of commission (EOCs) in response to fire-induced instrument failures, the interviewed Senior Reactor Operators (SROs) indicated that it was extremely unlikely that any single instrument failure would cause an EOC if the alarm response procedure were implemented with verbatim compliance. The communications and conduct of operations protocols at Waterford 3 require confirmation with redundant and/or diverse indications prior to changing the state of any equipment. There were no indicators or alarms identified during the simulator control panel walk downs for which the operator would be expected to take an unrecoverable immediate action that would otherwise be undesired.<br><br>The assessment of fire impacts on instruments combined with the interviews of operators, and the simulator walkdowns represents a process for identifying vulnerabilities (including single instrument) that is sufficient to meet the Standard requirements for ES-C2. This finding has no impact on quantified results and will have no impact on application of a 10 CFR 50.69 program. |
| **FQ-C1-01** | FQ-C1 | Not Met | Dependencies on combinations of HFEs have been utilized from the internal events PRA, but not addressed for specific combination on a scenario by scenario basis. This needs to be done to ensure all combinations have been addressed.<br><br>Provide additional justification for selection of the methodology use for combo events to address fire impacts. | The original finding was judged by the 2017 F&O closure teams as Open.<br><br>The NFPA 805 LAR, RAIs, and SE had additional details that were not in the PRA documentation. The Fire PRA HEPs and combinations have been properly developed following NUREG-1921. Any action with any instrument/control impact from the fire are failed (set to 1). Multipliers are applied to all other events/combos to account for increase failure rate during a fire. Any events were these are not applied were explicitly evacuated to document the basis for such treatment.<br><br>The update revision 6 Fire PRA documentation (Reference 39) has been revised to include more detail to ensure FQ-C1 is met with more thorough and detailed documentation.<br><br>This finding has no impact on quantified results and will have no impact on application of a 10 CFR 50.69 program. |

| Table A3-4 – **Fire PRA Open Peer Review Findings Assessed During F&O Closure Review)** | | | | |
|---|---|---|---|---|
| **Finding Number** | **Supporting Requirement(s)** | **Capability Category (CC)** | **Description** | **Disposition for 10 CFR 50.69** |
| **FSS-C1-01** | FSS-C1 | CC-I | Two points or a range of heat release values were not assigned to the ignition sources. A lower overall CDF will likely be achieved by using a two-point analysis or additional fire modeling to represent HRR profiles from ignition thru burnout and the corresponding probabilities of damage. | The 2017 F&O Closure review judged this finding as Open.<br><br>Waterford 3 applied a single Heat Release Rate (HRR) Modeling Treatment. This treatment offers a means for incorporation of fire modeling into the fire PRA in a manner that eliminates the need for separate scenario specific analyses which require significant effort for configuration control, review and update.<br><br>In reviewing Waterford 3 F&O resolution for the Fire PRA applications, the NRC graded this finding as follows: "The NRC staff finds that the resolution of the F&O, as described by the licensee in the LAR, would have a negligible effect on the evaluations relied upon to support fire risk evaluations and has no impact on the conclusions of the risk assessment and therefore the resolution of the F&O is acceptable or this application." (per NRC memo (Reference 18).<br><br>This modeling limitation would have no impact specific to the 10 CFR 50.69 application or program. |
| **FSS-D7-01** | FSS-D7 | CC-I | Fire detection and suppression system generic unavailability values were used, and outlier behavior and system unavailability were not specifically analyzed. To obtain a higher category, a specific WSES3 maintenance history review to assess outlier behavior is to be documented. This capability assessment is the same as for the Model of Record (MOR).<br><br>To move from CC-I to CC-II, specific WSES maintenance history review to assess outlier behavior is to be documented. | This was graded by the 2017 closure review team as partially resolved (with open documentation).<br><br>Plant specific suppression unavailability is applied to suppression data used in the Fire PRA. PSA-WF3-03-FSS-06 (Reference 42) uses plant specific data and generic data to develop failure values used for each detection/suppression system credited. Any plant specific outlier behavior is explicitly included in the values.<br><br>The F&O closure team judged this finding a documentation issue only. The Fire PRA documentation has been updated to address this finding. This finding has no impact on quantified results and will have no impact on application of a 10 CFR 50.69 program. |
| **FSS-E3-01** | FSS-E3 | CC-I | Evaluate the validity of other parametric uncertainty probability distributions (other than Ignition Frequency, which is the only quantitative uncertainty addressed).<br><br>Only qualitative discussions were provided with respect to the uncertainty intervals for most fire modeling parameters. Provide quantitative uncertainty intervals. | The 2017 F&O Closure review judged this finding as 'Open'.<br><br>Quantitative uncertainty intervals were generated and documented. The closure team determined that inputs to the uncertainty calculation were overly simplistic (error factors of 5 were applied to all ignition frequency inputs – actual error factor data was not applied). The closure team evaluated the uncertainty completed as insufficient to close the finding.<br><br>The 2019 Fire PRA update included detailed quantitative uncertainty evaluation for Fire PRA parameters. This includes |

| Table A3-4 – **Fire PRA Open Peer Review Findings Assessed During F&O Closure Review)** | | | | |
|---|---|---|---|---|
| **Finding Number** | **Supporting Requirement(s)** | **Capability Category (CC)** | **Description** | **Disposition for 10 CFR 50.69** |
| | | | | quantitative uncertainty intervals for fire frequencies, fire suppression factors, modeled circuit failure inputs, and credited recovery events. The updated uncertainty document (PSA-WF3-03-UNC-02, Reference 35) includes the necessary parametric uncertainty content to resolve this finding.<br><br>The documentation has been updated to address this finding. This finding has no impact on quantified results and will have no impact on application of a 10 CFR 50.69 program. |
| **FSS-F2-01** | FSS-F2 | CC-I | Criteria for structural collapse or non-collapse was not provided. Only judgment statements were provided, and these statements do not appear to reflect reality.<br><br>Re-perform the analysis to address the situation where a turbine building collapse occurs due to a large turbine lube oil fire. Revise the documents to eliminate the implication that failure of structural steel is not a credible event. | This was graded by the 2017 closure review team as partially resolved.<br><br>The closure review team assessed that the non-suppression of the structural collapse scenario is not sufficiently evaluated.<br><br>In the 2019 Fire PRA update:<br><br>The Turbine Generator Building (TGB) oil fire leading to collapse - manual suppression is not credited, since it is a fast-growing oil fire, but auto suppression is credited, since there is a dedicated deluge system for the TG Oil skid, FPM-5. The failure probability is based on actual plant data. This scenario fails all of the FPRA targets in the Turbine Building (except those in the Turbine Building Switchgear Room).<br><br>The model and documentation have been updated to address this finding. This finding has no impact on quantified results and will have no impact on application of a 10 CFR 50.69 program. |
| **FSS-H2-01** | FSS-H2 | CC-I | Section 4.0 (scenarios report) details damage criteria used in the Fire Scenarios. No cases of where plant specific thresholds or damage mechanisms were used.<br><br>Plant specific target damage evaluations that involve combinations of thermoset and thermoplastic cable have not been made. Only generic damage mechanisms have been used. | This was graded by the closure review team as partially resolved (with open documentation).<br><br>At the closure review, this finding remained open only due to documentation. The assessment of location and quantities for thermoset cables was thoroughly addressed in the NFPA 805 LAR, RAIs, and SER. It was not explicitly documented in the documents supporting the Fire PRA model. This limitation was noted by the review team and resulted in the finding only being partially closed.<br><br>The 2019 Fire PRA documentation contains the necessary documentation to fully resolve this finding. PSA-WF3-03-FSS-02 (Reference 40) and PSA-WF3-03-FSS-03 (Reference 41) calculate damage thresholds for both thermoplastic and thermoset |

| Table A3-4 – **Fire PRA Open Peer Review Findings Assessed During F&O Closure Review)** | | | | |
|---|---|---|---|---|
| **Finding Number** | **Supporting Requirement(s)** | **Capability Category (CC)** | **Description** | **Disposition for 10 CFR 50.69** |
| | | | | cables for various fire scenarios. Generic treatments are used for most scenarios. More detailed fire modeling is applied to the Relay Rooms. Different damage thresholds were also developed and applied for sensitive (solid state) electronics.<br><br>This has no impact on the 10 CFR 50.69 program. This is a documentation issue only and will have no impact on quantified results. |
| **FSS-H3-01** | FSS-H3 | Not Met | The basis for using the FDT tools over other fire modeling tools has not been provided. It is clear that the FDT tools have been V&V'd, (verification and validation) but their application to the specific scenarios involved in the analysis must also be documented.<br><br>The basis for using the FDT tools over other fire modeling tools has not been provided. | This was graded by the 2017 closure review team as partially resolved (with open documentation).<br><br>At the closure review, this finding remained open only due to documentation. Much of the details of the verification and validation for FDT is addressed in the NFPA 805 LAR, RAIs, and SER. At the time of the F&O Closure review some of these details were not explicitly in the documents supporting the Fire PRA model. This limitation was noted by the review team and resulted in the finding only being partially closed.<br><br>The necessary details have been added to the Fire PRA documentation as part of the 2019 Fire PRA model update. PRA-W3-03-FSS-02 (Reference 40) , documents the verification and validation (V&V) of the fire modeling tools used in the Waterford 3 fire PRA including fixed ignition sources, transient ignition sources, multicompartment scenarios, and the main control room analysis through a comparison to published parameter values related to the various fire modeling tools.<br><br>This has no impact on the 10 CFR 50.69 program. This is a documentation issue only and will have no impact on quantified results. |
| **HRA-A4-01** | HRA-A4 | Not Met | WSES3 Fire Probabilistic Risk Assessment, Quantification Model Preparation and Database Development, R0247070001.03, Rev. 0, Appendix A, Single HFE Screening Process Results, has a "Cue source/ Instrumentation" field that identifies for a number of records, the applicable procedure(s) that would be utilized to address the respective HFE. Additionally, Appendix D, "Detailed HRA for Selected HFEs," does identify for certain HFEs, that there were limited STA reviews of the certain aspects of the HFEs, namely the cues. However, documented interviews with Operations to support the use of these cues/instruments were not found. Report | This was graded by the closure review team as partially resolved (with open documentation).<br><br>At the time of the 2017 F&O Closure review, the report with the documented interviews was complete and available, the interview attachment was not included in the relevant Fire PRA report.<br><br>Fire PRA specific interview (for operator actions during fire events) have been conducted and documented. The interviews include discussion or procedures, cues, instrumentation, as well as the potential for fire impacts to the actions. These interviews are |

| Table A3-4 – **Fire PRA Open Peer Review Findings Assessed During F&O Closure Review)** | | | | |
|---|---|---|---|---|
| **Finding Number** | **Supporting Requirement(s)** | **Capability Category (CC)** | **Description** | **Disposition for 10 CFR 50.69** |
| | | | PRAW3-01-001S03, Rev. 1, Operator Interview Sheets document the PRA Internal Events HRA events that were reviewed with Operations and includes the name of the Operator interviewed. No equivalent Operator Interview sheets were located to address the impact of fire. | documented in Engineering Change EC 46718 (Documentation of Operator Interviews from Appendix E of PSA-W3-03-PRM).<br><br>This has no impact on the 10 CFR 50.69 program. This was judged a documentation issue. The documentation has been resolved to address the issue. This Finding has no impact on quantified PRA results or application of a 10 CFR 50.69 program. |
| **HRA-E1-01** | HRA-E1 | Not Met | A part of the documentation uses future tense and requires correction. The missing dependency analysis needs to be added as well as the documentation for operator and training interfaces.<br><br>Documentation issues with R0247070001.03, Revision 0, Section 5: Page 5-8 has a paragraph explaining that ex-MCR HFEs are set to true and then the risk significant HFEs are analyzed in more detail. If the HFE is set to true, then it would not show up in the cutsets. This paragraph needs a rewrite. Page 59 has a table explaining various treatments of HFEs in the model. Two of the columns conflict, one recommends a course of action and the resolution takes another course with no explanation of the differences. Also, there should be some discussion about thermal hydraulic analysis on any new sequences. | This was graded by the 2017 closure review team as partially resolved (with open documentation).<br><br>The assessment of internal events PRA HRA actions used in the FPRA is discussed in the Plant Response Model report (PSA-WF3-03-PRM,Reference 39). The impact of the fire on the action (instruments, increased stress) is also evaluated. The current/updated document provides a clearer description of the applied methodology for HFEs.<br><br>The finding was judged partially closed due to issues concerning documentation of the HEP dependency treatment for the fire PRA. The review team concluded through the referenced report, RAI responses, and review of the model recovery rule files that the treatment is acceptable. The open issue is that the details should all be in the report and review of RAI and recovery rule files should not be necessary to make such judgments.<br><br>The Fire PRA update in 2019 provides a more thorough documentation of HFE treatment. The assessment of internal events PRA HRA actions used in the FPRA is discussed in the Plant Response Model (PSA-WF3-03-PRM, Reference 39). The impact of the fire on the action (instruments, increased stress) is also evaluated. The current document provides a clearer description of the applied methodology for HFEs.<br><br>The 2019 Fire PRA update provides more thorough documentation of the issue involved in this Finding. This finding has no impact on quantified PRA results or application of a 10 CFR 50.69 program. |
| **IGN-A10-01** | IGN-A10 | CC-I | The Fire Frequency results presented in Table 4-1 are mean values only. Previous uncertainties associated with the Bin frequencies and the results of the Bayesian update are described. Uncertainties associated with the application of the weighting factors which multiply the Bin frequencies are not discussed anywhere. There is no | This was graded by the 2017 closure review team as 'Open'.<br><br>Quantitative uncertainty intervals were generated and documented. The closure team determined that inputs to the uncertainty calculations were overly simplistic (error factors of 5 were applied to all ignition frequency inputs – actual error factor |

| Table A3-4 – **Fire PRA Open Peer Review Findings Assessed During F&O Closure Review)** | | | | |
|---|---|---|---|---|
| **Finding Number** | **Supporting Requirement(s)** | **Capability Category (CC)** | **Description** | **Disposition for 10 CFR 50.69** |
| | | | discussion of uncertainties in this area in the final quantification report.<br><br>Documentation of sources of uncertainty covers uncertainties associated with NUREG/CR-6850 Bin elements, and uncertainties associated with the Bayesian update. No discussion on uncertainties associated with partitioning and weighting factor applications. The uncertainty analysis is incomplete. Provide either a numerical uncertainty analysis or qualitative discussion of other sources of uncertainty as required by the standard. | data was not applied). The closure team evaluated the uncertainty completed as insufficient to close the finding.<br><br>The 2019 Fire PRA update included detailed quantitative uncertainty evaluation for Fire PRA parameters. This includes quantitative uncertainty intervals for fire frequencies, fire suppression factors, modeled circuit failure inputs, and credited recovery events. The updated parametric uncertainty document (PSA-WF3-03-UNC-02, Reference 35) includes the necessary parametric uncertainty) includes the necessary parametric uncertainty content to resolve this finding.<br><br>A second uncertainty document, PSA-WF3-03-UNC-01 (Reference 34) documents sources of uncertainty and examines the impact the sources have on the model and results. The PSA-WF3-03-UNC-01 characterizes sources of uncertainty from all relevant Fire PRA tasks.<br><br>The documentation has been updated to address this finding. This finding has no impact on quantified results and will have no impact on application of a 10 CFR 50.69 program. |
| **UNC-A1-01** | UNC-A1 | Not Met | QU-E1 and QU-E2 requires identification of sources of model uncertainty and assumptions. In general, WSES had an assumption section in each report. However, a simple search on "assume" showed that there were many more assumptions than were listed in the assumption sections. At the CC-I level, QU-E3 requires estimation of the uncertainty interval of the overall CDF results. WSES does not provide an estimation of the uncertainty interval for CDF. QU-E4 requires that for each source of model uncertainty and related assumption identified in QU-E1 and QUE2, respectively; IDENTIFY how the PRA model is affected. WSES only identifies how the model is impacted for some of the assumptions and sources of uncertainty. The Uncertainty and Sensitivity Matrix in Appendix D of R0247070001.07. Per LE-F2, WSES should review LERF contributors for reasonableness (e.g., to assure excessive conservatisms have not skewed the results, level of plant specificity is appropriate for significant contributors, etc.). There is no evidence that such a review was performed. Additional explanation should be provided for the documented entries. The meaning of the confidence intervals for the different values (mean, 5th/95th, and | This was graded by the 2017 closure review team as 'Open'.<br><br>A detailed uncertainty evaluation for each task in the methodology is provided in the uncertainty report (PSA-WF3-03-UNC-01, Reference 34). The document provides qualitative and quantitative uncertainties with explanations for the associated analyses.<br><br>The 2019 Fire PRA update includes two documents dedicated to uncertainty. One document covers quantitative uncertainty and includes a much more thorough assessment of parametric uncertainty. Quantitative uncertainty intervals were generated and documented.<br><br>The 2019 Fire PRA update includes quantitative uncertainty intervals for fire frequencies, fire suppression factors, modeled circuit failure inputs, and credited recovery events. The updated parametric uncertainty document (PSA-WF3-03-UNC-02, Reference 35) includes the necessary parametric uncertainty content to resolve this finding. |

| Table A3-4 – **Fire PRA Open Peer Review Findings Assessed During F&O Closure Review)** | | | | |
|---|---|---|---|---|
| **Finding Number** | **Supporting Requirement(s)** | **Capability Category (CC)** | **Description** | **Disposition for 10 CFR 50.69** |
| | | | median) would not be obvious to most readers. It is suggested that this discussion be corrected in the next update of the documentation. The usefulness of the qualitative evaluation of modeling uncertainties could be significantly enhanced by additional comments regarding their potential impacts. EPRI 1026511 provides a tabulation of sources of modeling uncertainty associated with fire. There is no indication that this report was considered in identifying the sources discussed in the documentation. Because this has become a standard reference, and is a companion document to NUREG-1855, it would be worthwhile to check EPRI 1026511 to identify any additional sources that should be discussed. | A second uncertainty document, PSA-WF3-03-UNC-01 (Reference 34) documents sources of uncertainty and examines the impact the sources have on the model and results. This report (as the finding suggests) evaluates uncertainty topics from EPRI 1026511. The PSA-WF3-03-UNC-01 characterizes sources of uncertainty from all relevant Fire PRA tasks and examines the impact of assumptions made in those tasks.<br><br>The documentation has been updated to address this finding. This finding has no impact on quantified results and will have no impact on application of a 10 CFR 50.69 program. |
| Note – All findings in Table 2-4 are F&Os that remain open/partially open following a 2017 F&O Closure review. The original findings evaluated during the closure review were from a 2011 full scope fire peer review and two focused scope peer reviews conducted in September 2012 and May 2013. | | | | |

**Enclosure, Attachment 4**

**W3F1-2020-0047**

**External Hazards Screening**

# EXTERNAL HAZARDS SCREENING

| External Hazard | Screening Result | | |
|---|---|---|---|
| | **Screened?** (Y/N) | **Screening Criterion** (Note a) | **Comment** |
| Aircraft Impact | Y | PS3 | Aircraft hazards were evaluated at design stage of Waterford 3. FSAR section 2.2.3.5 states that the closest military base is the Naval Air Station in Belle Chasse, Louisiana which is approximately 30 miles east-southeast of Waterford 3. There are no training flights or bombing runs associated with this base in the vicinity of the nuclear plant nor is there any unique military aeronautical activity in the area of Waterford 3 that should affect the safety of the plant, so it is screened out from further consideration.<br><br>There are several airports within 10 miles of the plant, but only two (New Orleans International and Triche Airstrip were evaluated in detail (these met the traffic and proximity criteria). Potential aircraft strikes were calculated for the two unscreened airports and the conservative estimate resulted in damage frequency that was less than 1E-06/yr for safety related buildings. (Reference 26).<br><br>Aircraft hazard is not a design basis hazard event for the plant and this review using the most recent data confirms this conclusion. The bounding analysis satisfies Criterion PS3 from the table in Attachment 5. Therefore, aircraft hazard can be screened out from external event PRA for Waterford 3. |
| Avalanche | Y | C3 | Not applicable to the site because of climate and topography. |
| Biological Event | Y | C1 | There would be adequate warning for these events. Also note that the Mississippi River is not the Ultimate Heat Sink (UHS) for Waterford 3. |
| Coastal Erosion | Y | C3 | Waterford 3 site is inland along the Mississippi River. |
| Drought | Y | C1, C5 | There have been low levels in the Mississippi River; however, the ultimate heat sink consists of the dry and wet cooling towers and a 30-day supply of water stored in the wet cooling tower basins. In addition, the plant can receive water from on-site water sources as well as the Mississippi River. |

| External Hazard | Screening Result | | |
|---|---|---|---|
| | **Screened?** (Y/N) | **Screening Criterion** (Note a) | **Comment** |
| External Flooding | Y | PS1 | The Waterford 3 IPEEE found that external flooding events would cause no flooding of site safety-related structures, systems and components (SSCs). The updated evaluation examined: updates to plant data, flood data, new measures for risk management (including FLEX), for any new research on the topic (including those related to Fukushima), and new or updated plant procedures related to external flooding since the time the Waterford 3 IPEEE was published. The updated report confirmed the IPEEE conclusion that external flooding can be screened (Reference 26). |
| Extreme Wind or Tornado | Y | PS4 | Waterford 3 has been designed for extreme winds and tornado loadings that are substantially higher than the design basis events presently required. The SSCs are protected from tornado missiles using barriers with thicknesses exceeding the current requirements based on recent tornado hazard analysis. Therefore, the Waterford 3 design meets the supporting requirement C1 for Progressive Screening of External Events. It is concluded that the hazard event of extreme winds and tornadoes can be screened out from the PRA of Waterford 3. |
| Fog | Y | C4 | It affects frequency of occurrence of other hazards, e.g., highway accidents, aircraft landing and take-off accidents and is indirectly considered. Fog occurs on average 32 times per year, with most occurring from November through March in the site region (Reference 26). |
| Forest or Range Fire | Y | C1<br>C3 | The site/area landscaping and lack of forestation prevent such fires from posing a threat to Waterford 3.<br><br>There are no forests close enough to the control room. Therefore, the release of toxic combustion products from forest fire does not pose a hazard to control room personnel, nor will it cause thermal damage to the Waterford 3 safety-related structures. |
| Frost | Y | C1<br>C3 | Limited occurrence because of location and climate (southern Louisiana). Effects bounded by snow and ice. |
| Hail | Y | C1 | Hail may occur, but there are no openings in the walls or roofs of safety related buildings through which hail may enter and damage essential equipment. Tornado missile protection features, structural walls and roofs are adequate to withstand the impact of hail. |

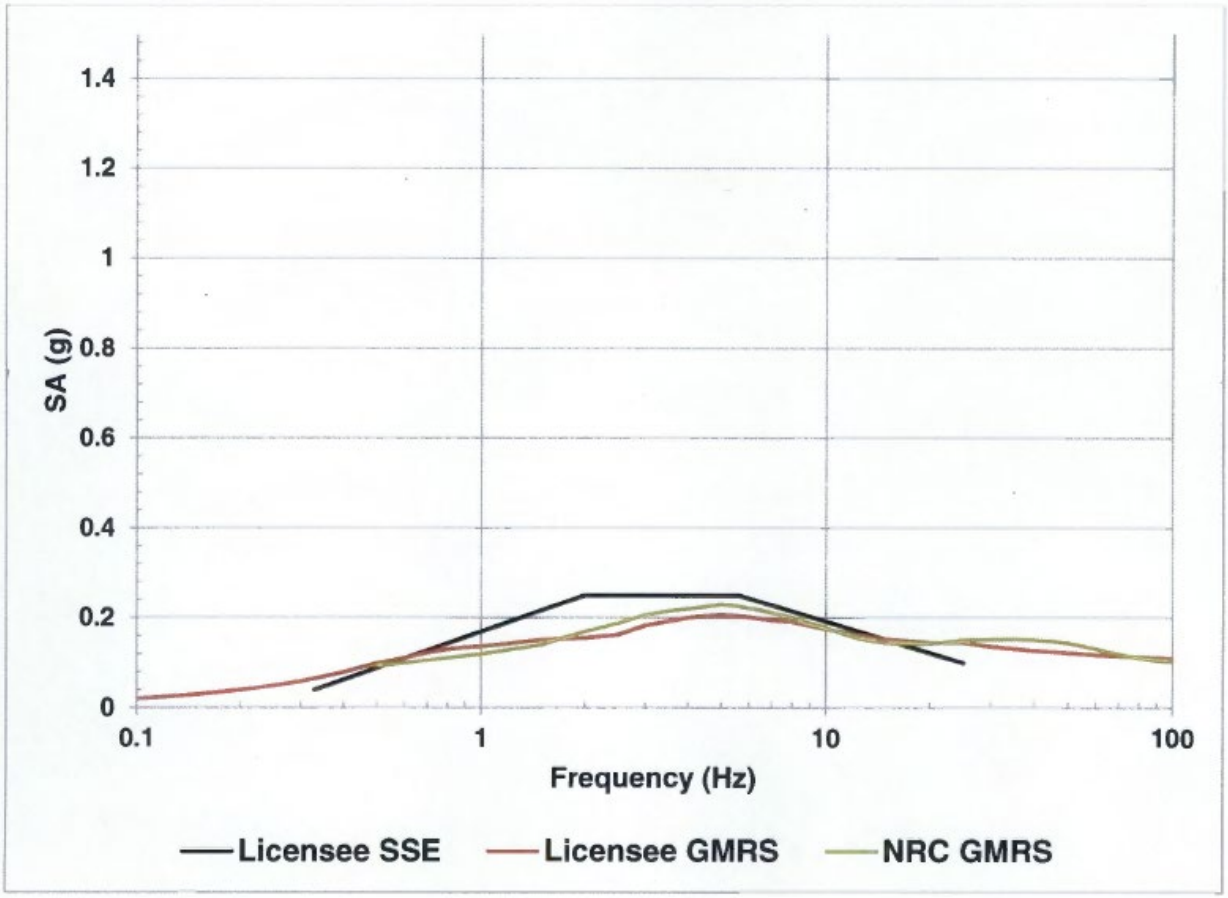
| External Hazard | Screening Result | | |
|---|---|---|---|
| | **Screened?** (Y/N) | **Screening Criterion** (Note a) | **Comment** |
| High Summer Temperature | Y | C1 | The plant is designed for this hazard. Associated plant trips have not occurred and are not expected.<br><br>The principal effects of such events would result in elevated Cooling Tower basin (UHS) temperatures which are monitored by station personnel. Should the basin temperature exceed the Technical Specification limit, an orderly shutdown would be initiated. These effects would take place slowly allowing time for orderly plant reductions, including shutdowns. |
| High Tide, Lake Level, or River Stage | Y | C4 | Included in external flooding analysis |
| Hurricane | Y | C1 | The plant is near the Gulf Coast; however, wind speeds generated during hurricanes and other storms are less intense and lower in magnitude than those generated by tornadoes. Thus, plant structures that are designed to satisfy the design criteria for tornadoes will also satisfy the design criteria for those events categorized as "high winds" in hurricanes.<br><br>Storm surge associated with hurricanes is covered in external flooding.<br><br>Hurricanes involve advanced notice and site procedures require shutdown prior to hurricane conditions at or near the site. |
| Ice Cover | Y | C3 | There have been only 3 significant glaze occurrences in the site's vicinity since the 1920's, glaze being a clear coating of ice containing some air pockets. The ice had entirely melted within less than 24 hours after the worst of these events and after only several hours for the other two. (Reference 26) Due to the relatively infrequent occurrence of ice cover, this event is screened out from further consideration. |
| Industrial or Military Facility Accident | Y | PS2<br>PS3 | Nearby facilities at the time of Operating License (OL) issuance which had the potential to affect Waterford 3 were identified and evaluated in FSAR Section 2.2. The nature and extent of activities conducted at nearby facilities, including the products and materials likely to be processed, stored, used, or transported have been identified and evaluated. Sufficient statistical data has been documented to establish a basis for evaluating the potential hazards to the plant. The potential hazards are of two types; (1) toxic gas hazards, and (2) fire and explosion hazards. These hazards can result from various manufacturing industries, pipelines, roads and railroads, and ship traffic in the Mississippi River.<br><br>The PRA-WF3-07-01 report (Reference 26) evaluated the following with regard to |

| External Hazard | Screening Result | | |
|---|---|---|---|
| | **Screened?** (Y/N) | **Screening Criterion** (Note a) | **Comment** |
| | | | industrial/military accidents:<br>• Identified Significant Changes Since OL Issuance<br>• Potential Fire/Explosion Hazard from Major Depots or Storage Areas<br>• Fire (offsite fire impacting the site)<br>• Explosion<br>• Toxic Gas Hazards<br><br>Based on the evaluations at Waterford 3 on storage and handling of toxic chemicals near the site, it is concluded that this hazard group does not pose a credible threat to Waterford 3. Therefore, the external hazard from industrial and military facilities accidents are screened out from the Waterford 3 PRA. |
| Internal Flooding | N | None | PRA is performed for this event. |
| Internal Fire | N | None | PRA is performed for this event. |
| Landslide | Y | C3 | Not applicable to the site because of topography. |
| Lightning | Y | C1 | Lightning strikes causing loss of offsite power or turbine trip are contributors to the initiating event frequencies for these events. However, other causes are also included. The impacts are no greater than already modeled in the internal events PRA. |
| Low Lake Level or River Stage | Y | C1<br>C5 | Waterford 3 does not rely on the Mississippi River for ultimate heat sink. Extreme low level of the Mississippi could lead to a plant shutdown but it would be a slow developing event and a controlled shutdown. |
| Low Winter Temperature | Y | C1 | Extended freezing temperatures are rare for southern Louisiana. The plant location, design, and operating procedures (site has procedures to prevent impact from low temperatures) prevent external low/winter temperatures from impacting the plant. |
| Meteorite or Satellite Impact | Y | C2 | This event has a very low frequency of occurrence for any site in the US. |
| Pipeline Accident | Y | PS4 | FSAR Section 2.2.3.1.3 has analyzed the hazard from pipeline accidents.<br><br>There are approximately 49 major pipelines within 2 miles of the Waterford 3 site (Reference 26). Products carried in these pipelines include natural gas, hydrogen, ammonia, LPG, ethane, gasoline, propane, and raw materials. These and other major pipelines carrying hazardous materials within a 2-mile radius of Waterford 3 are listed in PRA-WF3-07-01.<br><br>The assessment of pipeline accidents considered the pipe size, contents, flow rate of contents, and proximity to the site when considering accident impact. The assessment concluded that none of |

| External Hazard | Screening Result | | |
|---|---|---|---|
| | **Screened?** (Y/N) | **Screening Criterion** (Note a) | **Comment** |
| | | | the pipelines represented a significant hazard. Postulated damage from pipeline incidents is well within the design basis of the Waterford 3 safety related structures and is in fact bounded by previously analyzed explosive events. |
| Release of Chemicals in Onsite Storage | Y | C4 | FSAR Section 2.2.3.3 has analyzed the hazard from onsite storage of chemicals.<br><br>PRA-WF3-07-01 (Reference 26) lists the updated (2016) on-site sources of toxic chemicals at Waterford 3 and their disposition. Release of chemicals on site is not considered risk significant and is screened from further analysis for risk applications. |
| River Diversion | Y | C3 | The site is in southern Louisiana very near the mouth of the Mississippi where it meets the Gulf of Mexico. There is no historical or topographical evidence indicating that flow in the river can be diverted away from its present course. Due to the great width of the Mississippi and the relatively flat surrounding terrain, there is little possibility that natural or manmade obstructions could completely divert the flow away from its current path near the plant (especially not rapidly enough for the site not to respond as needed). River diversion will not impact Waterford 3. Note the Mississippi River is not used as UHS of Waterford 3. |
| Sand or Dust Storm | Y | C3 | This is not relevant for this region. See FSAR 2.1.1.1. |
| Seiche | Y | C3 | Lac des Allemands is about 5.5 miles southwest of the site and Lake Pontchartrain is about seven miles northeast of the site. There is no large body of water close to the site for this event. |
| Seismic Activity | N | None | See information in Section 3.2.3 of this application. |
| Snow | Y | C3 | Snow or freezing precipitation is not a serious concern for a power plant in the southern Louisiana area. Snowfall amounts in excess of two inches have only been recorded four times in the 100 years of data available prior to 1975. |
| Soil Shrink-Swell Consolidation | Y | C1<br>C5 | Site procedures are in place to monitor differential settlement (Reference 26). |
| Storm Surge | Y | C4 | Included in External Flooding analysis. |
| Toxic Gas | Y | C4 | Included in transportation accident, on-site chemical release, and industrial and military facilities accidents. |

| External Hazard | Screening Result | | |
|---|---|---|---|
| | **Screened?** (Y/N) | **Screening Criterion** (Note a) | **Comment** |
| Transportation Accident | Y | PS1 & PS4 | Transportation hazards in the vicinity of Waterford 3 which had the potential to affect Waterford 3 were re-evaluated in PSA-WF3-07-01 (Reference 26). All chemicals frequently shipped by truck, ship, railroad, or barge in the Waterford 3 vicinity, were analyzed. The results of the analysis showed that postulated radiological release (incapacitated operators leading to core damage) was not risk significant. Analysis of postulated accidents on the transportation routes has shown that they do not pose a credible threat to Waterford 3. Therefore, this hazard class can be screened out from the Waterford 3 PRA. |
| Tsunami | Y | C3 | Not applicable to the site because of location. |
| Turbine-Generated Missiles | Y | PS4 | The updated Waterford 3 External Events Hazard Report (Reference 26) examines the potential for high energy turbine generated missiles impacting the site. The report documents assessment of high pressure and low-pressure turbines for missile potential/likelihood. The report concludes that core damage resulting from turbine generated missiles is less than 1E-7 and can be screened from external event PRA consideration. |
| Volcanic Activity | Y | C3 | Not applicable to the site because of location. |
| Waves | Y | C4 | Waves associated with adjacent large bodies of water are not applicable to the site. Waves associated with external flooding are covered under that hazard. |
| Note a – See Attachment 5 for descriptions of the screening criteria.<br>Note b – The content of this table is included in Waterford 3 PRA Report - PSA-WF3-07-01 - Re-Examination of External Events Evaluation in the IPEEE (Reference 26). | | | |

**Figure 3.4-1 Comparison of the Staff's GMRS with Licensee's GMRS and the SSE for the Waterford site**




Note: Source of Figure above is Reference 13.

**Enclosure, Attachment 5**

**W3F1-2020-0047**

**Progressive Screening Approach for Addressing External Hazards**

**PROGRESSIVE SCREENING APPROACH FOR ADDRESSING EXTERNAL HAZARDS**

| Event Analysis | Criterion | Source |
|---|---|---|
| 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. (Note - Not used explicitly to screen. Used only to include within another event.) | NUREG/CR-2300 and ASME/ANS Standard RA-Sa-2009 |
| | C5. Event develops slowly, allowing adequate time to eliminate or mitigate the threat. | ASME/ANS Standard RA-Sa-2009 |
| Progressive Screening | PS1. Design basis hazard cannot cause a core damage accident. | ASME/ANS Standard RA-Sa-2009 |
| | PS2. Design basis for the event meets the criteria in the NRC 1975 Standard Review Plan (SRP). | NUREG-1407 and ASME/ANS Standard RA-Sa-2009 |
| | PS3. Design basis event mean frequency is < 1E-5/y and the mean conditional core damage probability is < 0.1. | NUREG-1407 as modified in ASME/ANS Standard RA-Sa-2009 |
| | PS4. Bounding mean CDF is < 1E-6/y. | NUREG-1407 and ASME/ANS Standard RA-Sa-2009 |
| 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 |

**Enclosure, Attachment 6**

**W3F1-2020-0047**

**Disposition of Key Assumptions/Sources of Uncertainty**

## DISPOSITION OF KEY ASSUMPTIONS/SOURCES OF UNCERTAINTY

The Waterford 3 PRA at power internal events model uncertainty document (Reference 32), internal flooding uncertainty document (Reference 33 and fire PRA uncertainty document (Reference 34) each contain a detailed, thorough evaluation of uncertainty for the model of record. Each report follows EPRI/NUREG guidance and considers a variety of potential sources of uncertainty. Key sources of uncertainty are identified, and relevant sensitivity cases are documented to examine the key sources of uncertainty and their impact on results.

The purpose of this attachment is to disposition the impact of uncertainty in the PRA models for the 10 CFR 50.69 Risk-Informed Application. The baseline internal events PRA model, the internal flooding model, and Fire PRA model each document assumptions and sources of uncertainty. These models and the associated documentation have all been reviewed during the model peer reviews. The completeness uncertainty associated with scope and level of detail are documented in the models but are only considered for their impact on a specific application. No specific issues of PRA completeness have been identified in the results of the internal events PRA, internal flood, and fire PRA peer reviews. The approach for this application is, therefore, to review these documents to identify the key assumptions and sources of uncertainty which may be directly relevant to the 10 CFR 50.69 Program calculations or categorization efforts, and to identify needed sensitivity analyses where appropriate.

The Waterford 3 PRA models are continuously maintained and periodically updated under the guidance of fleet procedures. This includes continuous identification, documentation and tracking of open issues (the site maintains a Model Change Request (MCR) database for identified model issues). Entergy PRA guidance requires periodic model update as well as self-assessments and peer reviews. Model issues ranging from MCRs to open peer review Findings will be reviewed during 10 CFR 50.69 categorization efforts to ensure the open issues and the potential impact they may have on the program are understood. This may include additional 10 CFR 50.69 specific sensitivity cases if they are judged necessary.

Note: As part of the required 10 CFR 50.69 PRA categorization, sensitivity cases directed by NEI 00-04 (Reference 1), the Waterford 3 PRA model human error and common cause basic events are increased to their 95th percentile and also decreased to their 5th percentile values; and maintenance unavailability terms are set to 0.0. These and any other relevant sensitivity insights and results are capable of driving a component and respective functions to an HSS categorization. The uncertainty of the PRA modeled HEPs and CCFs are thus explicitly accounted for in the 10 CFR 50.69 application. These sensitivity cases will be completed on all quantified models.

<u>Internal Events and Internal Flood Uncertainty</u>

The process used to identify at power internal events related PRA model uncertainties and their impact is described in the Waterford 3 PRA uncertainty documentation (Reference 32). The internal flooding PRA model uncertainty document, PSA-WF3-01-IF-SOU (Reference 33), contains details related to key assumptions and sources of uncertainty relevant to the flooding risk. In the uncertainty documentation, NUREG-1855 (Reference 21) and EPRI report 1016737 (Reference 22) were used to provide guidance for a structured process for addressing uncertainties in PRA inputs and results in the context of risk-informed decision-making. Appendix A of EPRI 1016737 is used as a template to document plant-specific issue

characterization and assessments to fully satisfy the related supporting requirements. Uncertainty considerations for each model element (success criteria, human actions, data, etc.) were also documented to ensure a comprehensive evaluation of PRA uncertainty was completed.

The parametric uncertainty analysis for the Waterford 3 at power internal events model is documented in Reference 32, and Reference 33 contains the parametric flood uncertainty evaluation. The parametric uncertainty analyses address the State-of Knowledge Correlation (SOKC) by the use of system level type codes for basic events. This applies the same variability of all components of that type within a system during the analysis. The parametric uncertainty analyses in the PRA quantification model and documentation demonstrate that the point estimate mean values provide a close representation of the propagated mean values reflecting SOKC and the propagated mean total CDF and LERF values were confirmed to meet RG 1.174 Revision 3 (Reference 20).

Table 6-1 evaluated key assumptions and sources of uncertainty from the baseline internal events and internal flooding models and assesses the impact of the item(s) on application of a 10 CFR 50.69 program.

| Table 6-1 Internal Events / Internal Flooding PRA Assumptions & Sources of Uncertainty | | |
|---|---|---|
| **Assumption/ Uncertainty** | **Discussion** | **10 CFR 50.69 Disposition** |
| The model has been updated to include credit for FLEX equipment. The added FLEX modeling was the subject of a sensitivity evaluation to compare results with and without the added FLEX equipment. | The uncertainty report documents a sensitivity evaluation for the added FLEX changes. The sensitivity case shows that FLEX has an impact and reduces Station Blackout (SBO) contribution.<br><br>While this is documented in the uncertainty and sensitivity documentation, it is not considered a source of uncertainty. This was a sensitivity evaluation to gauge the impact of a change to the model. This was a necessary change to accurately update the model to match plant response and the potential use of FLEX equipment and strategies. | This does not represent a key source of uncertainty in the 10 CFR 50.69 application.<br><br>The Revision 6 Waterford 3 Uncertainty and Sensitivity Analysis contained a sensitivity on FLEX equipment. This sensitivity examined the impact the FLEX changes had on the model. As the current model reflects the as-built, as-operated plant, this is not considered a source of uncertainty. |
| Environmental impacts on initiating events (for example, intake, offsite power) | Local environmental impacts can increase or decrease the frequency of some initiators. An alignment for severe weather is included for Waterford 3 that addresses the impact of severe weather on both Loss of Offsite Power (LOOP) frequency and recovery.<br><br>The other environmental impact applicable for Waterford 3 is extremely high temperatures. Waterford 3 has Technical Specifications that limit plant operation with excessive temperature in the ultimate heat sink. Forced shutdowns due to high temperature would be included in the reactor trip data. | This was noted as a source of model uncertainty for the Revision 6 model update but should not impact the 10 CFR 50.69 application.<br><br>The environmental impacts can impact the online risk monitor or influence changes to IE frequency. However, 10 CFR 50.69 risk evaluations will use nominal conditions for assessing risk on equipment and systems. |
| Credit for repair and Recovery<br><br>This was noted as a source of uncertainty in both the Success Criteria and System Analysis sections. | Waterford 3 does not credit repair of any equipment. Off-site power recoveries are based on standard practices.<br><br>Not crediting equipment recovery was noted as a source of uncertainty tied to Success Criteria. Applying recoveries or completing more detailed off-site power recoveries could lower risk results.<br><br>A sensitivity case for EDG recovery is included in the quantification/sensitivity documentation. | No significant impact on the 10 CFR 50.69 application. Conservative treatment is more likely to result in additional HSS SSCs than to mask other results. |

**Table 6-1 Internal Events / Internal Flooding PRA Assumptions & Sources of Uncertainty**

| Assumption/ Uncertainty | Discussion | 10 CFR 50.69 Disposition |
|---|---|---|
| Several bounding inputs and conditions for LERF analysis were identified as potential sources of uncertainty<br>• Use of generic value for SG age/wear<br>• In-vessel core melt impact on TI-SGTR, and<br>• Credit for scrubbing | Bounding conditions for the Waterford 3 LERF model include induced SGTRs and credit for scrubbing.<br><br>Several sensitivity evaluations were included in the LERF documentation to evaluate the impact the bounding and potentially uncertain inputs have on LERF results. For example, average SG wear models were used even as Waterford 3 recently replaced SGs.<br><br>Sensitivity evaluations are documented in the LERF analysis for the impact of credited scrubbing, as well as in-vessel core melt on TI-SGTR | Since the treatment of induced-SGTR and scrubbing are primarily a phenomenological uncertainty issue for LERF, this is not expected to have any impact on the 10 CFR 50.69 categorization.<br><br>All noted issues are treated conservatively in the LERF model and are not expected to uniquely impact 10 CFR 50.69 categorization. |
| Human actions credited during severe accident conditions | The Waterford 3 LERF model contains three operator actions after core damage: "bump the pump," late depressurization, and LOOP recovery.<br><br>Due to the uncertain nature of post-accident/post core damage conditions, credit for these actions was identified as a source of uncertainty. A sensitivity case is included in the documentation with no post core damage human actions credited. | Treatment of post core damage human actions is not expected to have any impact unique to the 10 CFR 50.69 categorization.<br><br>All credited human actions are already subject to NEI guidance directed sensitivity cases. |
| Maintenance/operational activities (for example, switchyard work, system testing) | Waterford 3 includes numerous alignment flags that can determine the impact of maintenance or operational activities on risk. In addition, the risk associated with different types of switchyard work can also be evaluated. | This was noted in the PRA revision 6 documentation as a potential uncertainty candidate. However, 10 CFR 50.69 risk evaluations will use nominal conditions for assessing risk on equipment and systems. |
| Model may over-estimate contribution of pressure induced SGTR (PI-SGTR) to LERF. | Conservative criteria in LERF modeling may cause over estimation of LERF risk. In modeling, induced SGTR – bounding conditions were applied.<br><br>The PI-SGTR and TI-SGTR values used the average wear models for the steam generator but the steam generators were recently replaced (the replacement Waterford 3 SGs have less wear than industry average). | No significant impact on a 10 CFR 50.69 application. Conservative treatment is more likely to result in additional HSS SSCs than to mask other results. |

| Table 6-1 Internal Events / Internal Flooding PRA Assumptions & Sources of Uncertainty | | |
|---|---|---|
| **Assumption/ Uncertainty** | **Discussion** | **10 CFR 50.69 Disposition** |
| *The Following Entries are Related to Internal Flooding:* | | |
| The information shown on the walkdown data sheets in the original internal flooding analysis was assumed to be correct unless changed in the Fire PRA Walkdown Notebook. The legacy information in the initial flooding walkdown was spot-checked to verify accuracy. | Different pipe lengths would directly impact the initiating event frequencies which in turn would affect the core damage frequency. For risk-significant flooding events, length data was based on drawing information or confirmed during walkdowns. Therefore, the overall effect on the base model is expected to be small. | No significant impact on the 10 CFR 50.69 application. Risk significant scenarios were checked and significant errors/discrepancies between actual and documented pipe lengths are not expected. |
| For areas where fluid drains away quickly, sealed penetrations are assumed to be effective at preventing propagation between areas such that the propagation would not directly impinge on equipment in the adjoining area and result in equipment failure. For example, the penetration seals will prevent spray and splash impacts from causing equipment failures in the adjoining area. Generic Letter 86-10 specifies that fire barriers must be capable of withstanding significant spray and splash after being exposed to fire. For areas where fluids drain away quickly, use of fire barriers is assumed to prevent propagation for the time needed for the fluid to drain away. | This assumption is reasonable and backed by regulatory requirements. It is also an industry consensus approach. Penetrations/barriers with a specific concern would be addressed on an individual basis. | This was noted in the document at a key assumption, but no sensitivity cases were developed as the treatment was judged to be acceptable and not a source of significant uncertainty.<br><br>This is a flood specific issue and is not expected to impact application of a 10 CFR 50.69 program. |
| Flood-induced failure of AOVs involves the valve operator's loss of function but would also involve the AOV failing to its designed fail position. However, if the fail-safe position is the desired position for the PRA, it is assumed that the valve fails as-is. | Due to multiple potential flood-induced failure modes (e.g., valve shorts out, instrument air supply line fails, water affects valve's diaphragm, etc.), it is not guaranteed that AOV will always fail to its fail-safe position. Therefore, credit for fail-safe is not taken. This assumption is conservative, but reasonable. The overall effect on the base model is expected to be small. | No significant impact on the 10 CFR 50.69 application. Conservative treatment is more likely to result in additional high safety SSCs than to mask other results. |
| It is considered improbable that flooding events with small flow rates will spray enough PRA related equipment housed in the affected zone to initiate a PRA transient. Therefore, the smallest discharge rate from any break is assumed to be 25 gpm unless noted otherwise [for specified spray scenarios]. | The selection of 25 gpm is based on engineering judgement, it is about the same flow rate as three or four household garden hoses and is considered reasonable.<br><br>This was identified in as a key assumption. This assumption is reasonable and expected to be conservative. The impact on the base model is expected to be small. | No significant impact on the 10 CFR 50.69 application. Conservative treatment is more likely to result in additional high safety SSCs than to mask other results. |

Fire PRA (FPRA) Model:

The process used to identify uncertainties and their impact is described for the fire PRA in the Waterford 3 FPRA uncertainty documentation (Reference 34).

The evaluation examines sources of uncertainty for each of the FPRA development follows NUREG-1855 (Reference 21). The Fire PRA Uncertainty Report contains discussions on topics outlined in from EPRI report 1026511 Appendix B (Reference 23) arranged by tasks from NUREG/CR-6850 (Reference 25). Key sources of uncertainty are noted, and sensitivity cases are completed to evaluate them.

The WF3 Fire PRA Sensitivity and Uncertainty Report is a thorough and comprehensive assessment of uncertainty. The report evaluates over seventy uncertainty topics and documents how each topic is addressed in the Waterford 3 Fire PRA model.

The parametric uncertainty analysis for the Waterford 3 Fire PRA is provided in the uncertainty documentation. The parametric uncertainty analysis addresses the State-of-Knowledge Correlation (SOKC) by the use of system level type codes for basic events, and additional SOKC factors included in quantitation. This applies the same variability of all components of that type within a system during the analysis. The parametric uncertainty analyses in the PRA quantification model/model documentation demonstrate that the point estimate mean values provide a close representation of the propagated mean values reflecting SOKC and the propagated mean total CDF and LERF values were confirmed to meet RG 1.174 Revision 3 (Reference 20).

The Fire PRA uncertainty document identifies key assumptions and sources of uncertainty relevant to development and quantification of the fire models. The items identified in that document are included in Table 6-2 below. The table includes items judged to be key assumptions and sources of uncertainty relevant to the fire PRA. Table 6-2 assess whether the key fire PRA assumptions and uncertainty topics are specifically relevant to the use of that model for the 10 CFR 50.69 application.

As noted in the uncertainty/sensitivity documentation, the WF3 Fire PRA analysis is believed to represent a somewhat conservative estimation of fire risk, within the constraints of the requirements for a model acceptable for the NFPA-805 program. As the model is somewhat conservative, its application for a 10 CFR 50.69 program will likely slightly bias results toward more HSS classification of components/system. The evaluation of sources of uncertainty in the FPRA are documented in the table below including consideration for impact on a 10 CFR 50.69 application.

The Fire PRA uncertainty document identifies key assumptions and sources of uncertainty relevant to development and quantification of the fire models. The items identified in that document are included in Table 6-2 below. The table includes items judged to be key assumptions and sources of uncertainty relevant to the fire PRA. Table 6-2 assess whether the key fire PRA assumptions and uncertainty topics are specifically relevant to the use of that model for the 10 CFR 50.69 application.

| Table 6-2 Fire PRA **Sources of Model Uncertainty** | | |
|---|---|---|
| **Uncertainty Topic** | **Sources of Uncertainty** | **10 CFR 50.69 Disposition** |
| Scope of equipment credited in fire PRA model | Lack of credit for some systems could mask the risk associated with those systems in some applications. Additionally, that same lack of credit could overestimate the importance of other credited systems.<br><br>A sensitivity case was completed to evaluate this. The Temporary Emergency Diesel Generator (TEDG) credit was adjusted. The TEDG was made permanent during a plant modification and was added to the PRA model in the last internal events update. There is some uncertainty with regard to how quickly the TEDG would be used in the event of a fire. For this sensitivity, it is assumed the TEDG is not available. | Categorization evaluations for the Emergency Diesel Generator system, if selected for categorization, will include sensitivity studies associated with the TEDG.<br><br>Besides the EDG/TEDG related systems, this topic is not a source of uncertainty for the 10 CFR 50.69 program (the sensitivity case resulted in less than 1% change in fire CDF and LERF). |
| Exclusion of certain systems due to lack of cable data | Lack of credit for some systems (systems with limited cable data – all assumed failed) could mask the risk associated with those systems in some applications. Additionally, that same lack of credit could overestimate the importance of other credited systems.<br><br>A sensitivity case was completed to evaluate this uncertainty. The model was evaluated with credit for Unlocated (assumed failed) Equipment. In the case run, the same equipment was assumed available in all locations except Turbine Generator Building (TGB). | The current approach used (assume equipment lacking detailed cable data is failed) will result in conservative evaluations. This conservativism would tend to result in additional SSCs being categorized as HSS in the 10 CFR 50.69 categorization process. |
| Development of fire frequencies for each fire area and ignition source | Present NUREG/CR-6850 results in different fire frequencies for the same equipment in different plants. For example, older BWRs with less equipment than a new PWR may result in a factor of 2 higher fire frequencies for pumps or electrical equipment. This is a form of parameter uncertainty. The Waterford 3 FPRA uses the ignition frequencies from EPRI Supplement 1 to NUREG/CR-6850. The source of data for the ignition frequencies is a source of uncertainty.<br><br>A sensitivity case was completed to evaluate this uncertainty. The model was run with ignition frequencies from different sources (NUREG/ CR-6850 vs. EPRI Supplement 1). | This is a fire frequency specific issue. This will not have an appreciable impact on the 10 CFR 50.69 categorization for Waterford 3 plant systems. |

| Table 6-2 Fire PRA **Sources of Model Uncertainty** | | |
|---|---|---|
| **Uncertainty Topic** | **Sources of Uncertainty** | **10 CFR 50.69 Disposition** |
| Credit for fire wrap | No credit is taken for fire wrap (qualified 3M wrap) in the Waterford 3 FPRA.<br><br>A sensitivity case was completed to evaluate this uncertainty. Credit for Wrap in Risk Significant locations where wrap exists but is not credited was evaluated (it assumed the wrap prevents failure of FPRA targets that are wrapped). | The inclusion/exclusion of fire wrap has a negligible impact on the overall results. Due to the small impact demonstrated by sensitivity case, the uncertainty sensitivity cases associated with this model uncertainty is negligible and not relevant for the 10 CFR 50.69 application. |
| Treatment of permissive signals, interlocks, and associated logic | In the Waterford 3 FPRA cable selection, the majority of the circuit analysis is performed using detailed and conservative safe shutdown analysis. Any additional cable selection is performed in a similar manner. Automatic actuation logic signals are modeled as separate pseudo-components with their own cable selection. Thus, associated circuits are accounted for in the cable selection. Some component specific changes were made in fire modeling/quantification (with documented basis for each decision). Due to the special treatment applied to certain signal cables. This treatment was identified as a source of modeling uncertainty.<br><br>Two sensitivity cases were documented to examine this uncertainty. In one case, the cables with special treatment were assumed to always fail. In the other case, the same cables were assumed to never fail. | The sensitivity cases were performed to measure the risk associated with the treatment of these cables that in select fire scenarios. Although the sensitivity shows a small impact on Fire CDF and Fire LERF, the uncertainty document concluded that the modeling treatment applied to these cables is realistic and preferred to alternative modeling treatments.<br><br>This is not expected to have any impact on the 10 CFR 50.69 categorization for Waterford 3 plant systems. |
| Circuit failure probabilities | Failure probabilities utilized in circuit failure analysis could be a form of parameter uncertainty, but the choice of the representative set of values is a form of model uncertainty. The Waterford 3 FPRA utilizes circuit failure probabilities from NUREG/CR-7150. The aggregate failure probabilities are used, which combines intra and inter cable faults, and is specific to the type of circuit and material type, e.g., single break control circuit, thermoset cables, ungrounded DC Solenoid Operated Valves (SOVs).<br><br>A sensitivity case was completed to examine this uncertainty. In the | The use of hot short failure probability given fire induced failure probability is based on fire test data and associated consensus methodology published in NUREG-7150, Volume 2. Based on a review of the assumptions and potential uncertainty related to this element, it is concluded that the methodology for the Circuit Failure Mode Likelihood Analysis task does not introduce any epistemic uncertainties that would affect a 10 CFR 50.69 program. |

| Table 6-2 Fire PRA **Sources of Model Uncertainty** | | |
|---|---|---|
| **Uncertainty Topic** | **Sources of Uncertainty** | **10 CFR 50.69 Disposition** |
| | sensitivity, all Circuit Failure Mode Likelihood (CFMLA) values were set to 1.0, i.e., no credit for hot short probabilities. | |
| Availability of power for spurious operations after initial cable failure | In the Waterford 3 FPRA, a simplifying assumption is made that the power is available to a cable damaged by a fire and thus can spuriously operate, when in fact the fire damage may cause the power supply to be interrupted. Power is assumed to be initially available to allow spurious operation to occur. This is a conservative assumption. An exception to this modeling assumption is in the Turbine Building, if power is lost to the 6.9kV switchgears, the RCPs are assumed to be tripped and cannot spuriously start or fail to be tripped by the operators, despite a spurious failure of the RCP control cables.<br><br>A sensitivity case was completed to examine this uncertainty. The case removes this treatment for the RCPs (they can spuriously start/operate with loss of 6.9kV power). | This is a fire PRA specific issue and is limited to the RCPs (components not likely to be included in 10 CFR 50.69 categorization). This is not expected to have any impact on the 10 CFR 50.69 categorization for Waterford 3 plant systems. |
| Credit for detection and suppression | In the Waterford 3 FPRA, manual suppression is not credited for oil fire scenarios, since it was determined that oil fires damage targets quickly, and it is assumed that target damage occurs prior to the fire brigade responding. For electrical and cable fires, a bounding time to damage is determined that is applicable to all such fire scenarios, and this time to damage is used to establish time available to credit manual suppression utilizing generic non-suppression probabilities, as provided in FAQ 08-0050 in Supplement 1 of NUREG/CR-6850 (Reference 54).<br><br>Several Sensitivity cases were run to examine the uncertainty and model sensitivity to credited detection and suppression. The following cases were completed.<br><br>• Assume failure of all automatic suppression systems (no credit for automatic suppression. | This is a fire detection/suppression specific issue. This is not expected to have any impact on the 10 CFR 50.69 categorization for Waterford 3 plant systems. |

| Table 6-2 Fire PRA **Sources of Model Uncertainty** | | |
|---|---|---|
| **Uncertainty Topic** | **Sources of Uncertainty** | **10 CFR 50.69 Disposition** |
| | • Credit automatic suppression in the general areas of the Turbine Building, PAU TGB.<br>• A case that assumes failure of all automatic detection systems and thus manual suppression in all areas that are not continuously manned, such as the MCR.<br>• A case was run to examine manual detection/suppression. The case assumed failure of all manual suppression by the dedicated fire watch in hot work scenarios, as well as manual suppression in the MCR and other continuously manned locations. | |
| Effectiveness of passive fire barriers between compartments | The fire barrier failure probabilities are based on the partitioning elements in the barrier. In the Waterford 3 FPRA, the failure probabilities for passive fire barrier partitioning elements are based on the generic values listed in NUREG/CR-6850 Appendix A. The barrier failure probability for the barrier is the sum of the partitioning elements applicable to that barrier.<br><br>A couple of sensitivity cases were run to examine uncertainty associated with credited barrier failure probabilities. In one case it was assumed all barriers fail with a probability of 1.0 (i.e., they are not effective due to a door being propped open, a penetration not being filled after maintenance, etc.).<br><br>In a second case, barrier failure probabilities were reduced. It was assumed all barriers are better by a factor of 10 (i.e., they are less conservatively modeled, unless the failure is currently 1.0). | This is a fire barrier specific issue. This is not expected to have any impact on the 10 CFR 50.69 categorization for Waterford 3 plant systems. |
| Treatment of structural steel failures | The potential impacts of structural steel failures were considered and assessed at Waterford 3, and it was determined that the TGB is the only PAU noted to contain potential high-hazard fire sources. All oil fire scenarios in the TGB were evaluated, and those that could possibly impact | This is a fire specific issue associated with large fires in the Turbine Building. This is not expected to have any impact on the 10 CFR 50.69 categorization for Waterford 3 plant systems. It is also a conservative treatment and any impacts would slightly bias systems/components more toward HSS classification. |

| Table 6-2 Fire PRA **Sources of Model Uncertainty** | | |
|---|---|---|
| **Uncertainty Topic** | **Sources of Uncertainty** | **10 CFR 50.69 Disposition** |
| | a structural member were assumed to fail the entire TGB outside the Switchgear Room. This assumption could contain a large amount of conservatism.<br><br>A sensitivity case was completed to examine this uncertainty.<br><br>A case was run to reduce conservatism with structural collapse fires in the turbine building. Some hydrogen fires and oil fires were adjusted to only impact the source of the fire and nearby targets, but not collapse the Turbine Building. | |
| Treatment of fire-induced instrument failures | For all operator actions credited in the Waterford 3 FPRA, there is a discussion of the cues and/or required instrumentation. Some actions model the required instrumentation. For other operator actions, redundancy and diversity of instruments is credited as a means to assume that sufficient indication exists. In this case, the assumption that it does not need to be modeled becomes a source of model uncertainty.<br><br>A sensitivity case was completed to examine this uncertainty.<br><br>Indication for the Condensate Storage Pool (CSP) Makeup or Alternate EFW Suction are among the most important indications for operators. This is because the system window is several hours long, it is assumed alternate indications for CSP level could be used as an operator cue; however, if alternate indication is not available or provides false readings, these actions would be hindered. This sensitivity case assumes failure probabilities of CSP makeup and alternate CSP suction are increased by factor of 10 (this includes relevant combination HFEs). | The Waterford 3 Fire PRA model is based on industry consensus modeling approaches for its HEP calculations, so this is not considered a significant source of uncertainty. As directed by NEI 00-04, human error basic events are increased to their 95th percentile and also decreased to their 5th percentile values as part of the required 10 CFR 50.69 PRA categorization sensitivity cases. These results are capable of driving a component and respective functions HSS and, therefore, the uncertainty of the Fire PRA modeled HEPs are accounted for in the 10 CFR 50.69 application. |
| HEP Methodology | The basis for the HEP methodology utilized needs to be consistent with the internal events PRA standard requirements for HRA. Model uncertainty exists on the actual methodology utilized; this is recognized as a generic source of | As directed by NEI 00-04, human error basic events are increased to their 95th percentile and also decreased to their 5th percentile values as part of the required 10 CFR 50.69 PRA categorization sensitivity cases. These results are capable of driving a component and |

| Table 6-2 Fire PRA **Sources of Model Uncertainty** | | |
|---|---|---|
| **Uncertainty Topic** | **Sources of Uncertainty** | **10 CFR 50.69 Disposition** |
| | model uncertainty. The Waterford 3 FPRA utilized EPRI HRAC per the guidance in NUREG-1921 (Reference 24) in the development of fire-specific HEP values.<br><br>Several Fire HEP related sensitivity cases were run to examine relevant uncertainties. The cases included:<br><br>A case examining - Manual Control of EFW Flow (Long term ex-MCR actions are assumed to have the same failure probability as the internal events model; however, if these actions would be hindered by the fire in some way, the failure probabilities would be higher; assume failure probabilities of these actions are increased by a factor of 10, including relevant combination HFEs.)<br><br>A case examining - Manual Actuation of EFAS if Auto-Actuation Fails (This is the most risk-significant single operator action; this is a simple action that is assumed to be a factor of 10 worse than the internal events probability, per the guidance in NUREG-1921; however, since this is the most important action and is a very simple action. The case assumes that the probability is equal to that calculated for internal events and reduces the relevant action and combination HFEs by a factor of 10.)<br><br>A case examining - Tripping of the RCPs Following Loss of Seal Cooling (Combination of tripping from the MCR, which is assumed to be a factor of 10 worse than the internal events value, and locally tripping the breaker in TGB, which has a detailed fire-specific calculation). This action is known by the operators to be an important action and is very simple. This case reduces the failures associated with tripping the RCPs (and relevant HFE combination events) by a factor of 10. | respective functions HSS and, therefore, the uncertainty of the Fire PRA modeled HEPs are accounted for in the 10 CFR 50.69 application. |