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10 CFR 50.90
10 CFR 50.69

July 30, 2021

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

James A. FitzPatrick Nuclear Power Plant
Renewed Facility Operating License No. DPR-59
NRC Docket No. 50-333

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

In accordance with the provisions of 10 CFR 50.69 and 10 CFR 50.90, Exelon Generation Company, LLC (Exelon) is requesting an amendment to the license of James A. FitzPatrick (JAF) Nuclear Power Plant.

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

The enclosure to this letter provides the basis for the proposed change to the JAF 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 PRA models described within this license amendment request (LAR) are the same as those described within the Exelon submittal of the LAR dated July 30, 2021, "License Amendment Request to Revise Technical Specifications to Adopt Risk Informed Completion Times TSTF-505, Revision 2, 'Provide Risk-Informed Extended Completion Times – RITSTF Initiative 4b,' " (ML21211A053). Exelon requests that the NRC conduct their review of the PRA technical adequacy details for this application in coordination with the review of the application currently in-process. This would reduce the number of Exelon and NRC resources necessary to complete the review of the applications. This request should not be considered a linked requested licensing action (RLA), as the details of the PRA models in each LAR are complete which will allow the NRC staff to independently review and approve each LAR on their own merits without regard to the results from the review of the other.

Exelon requests approval of the proposed license amendment by July 30, 2022, with the amendment being implemented within 60 days following NRC approval.

In accordance with 10 CFR 50.9, a copy of this application, with attachments, is being provided to the designated New York State Official.

Should you have any questions concerning this submittal, please contact Ron Reynolds at (610) 765-5247.

I declare under penalty of perjury that the foregoing is true and correct. Executed on this 30th day of July 2021.

Respectfully,

David T. Gudger

David T. Gudger
Senior Manager - Licensing
Exelon Generation Company, LLC

Enclosure: Evaluation of the Proposed Change

cc: USNRC Region I, Regional Administrator w/ attachments
USNRC Project Manager, FitzPatrick "
USNRC Senior Resident Inspector, FitzPatrick "
A. L. Peterson, NYSERDA "

Enclosure
Evaluation of the Proposed Change

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1 SUMMARY DESCRIPTION

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

2 DETAILED DESCRIPTION

2.1 CURRENT REGULATORY REQUIREMENTS

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

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

2.2 REASON FOR PROPOSED CHANGE

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

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

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

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

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

2.3 DESCRIPTION OF THE PROPOSED CHANGE

Exelon proposes the addition of the following condition to the renewed operating license of JAF to document the NRC's approval of the use 10 CFR 50.69.

> Exelon is approved to implement 10 CFR 50.69 using the processes for categorization of Risk-Informed Safety Class (RISC)-1, RISC-2, RISC-3, and RISC-4 Structures, Systems, and Components (SSCs) 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 and non-Class SSCs and their associated supports; the results of the non-PRA evaluations that are based on the IPEEE Screening Assessment for External Hazards updated using the external hazard screening significance process identified in ASME/ANS PRA Standard RA-Sa-2009 for other external hazards except seismic; and the alternative seismic approach as described in Exelon's submittal letter dated July 30, 2021, and all its subsequent associated supplements 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 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.

The PRA models described within this license amendment request (LAR) are the same as those described within the Exelon submittal of the LAR dated July 30, 2021, "License Amendment Request to Revise Technical Specifications to Adopt Risk Informed Completion Times TSTF-505, Revision 2, 'Provide Risk-Informed Extended Completion Times – RITSTF Initiative 4b,' " (ML21211A053).

Exelon requests that the NRC conduct their review of the PRA technical adequacy details for this application in coordination with the review of the application currently in-process. This would reduce the number of Exelon and NRC resources necessary to complete the review of the applications. This request should not be considered a linked requested licensing action (RLA), as the details of the PRA models in each LAR are complete which will allow the NRC staff to independently review and approve each LAR on their own merits without regard to the results from the review of the other.

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

3.1.1 Overall Categorization Process

Exelon will implement the risk categorization process in accordance with NEI 00-04, Revision 0, as endorsed by 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.

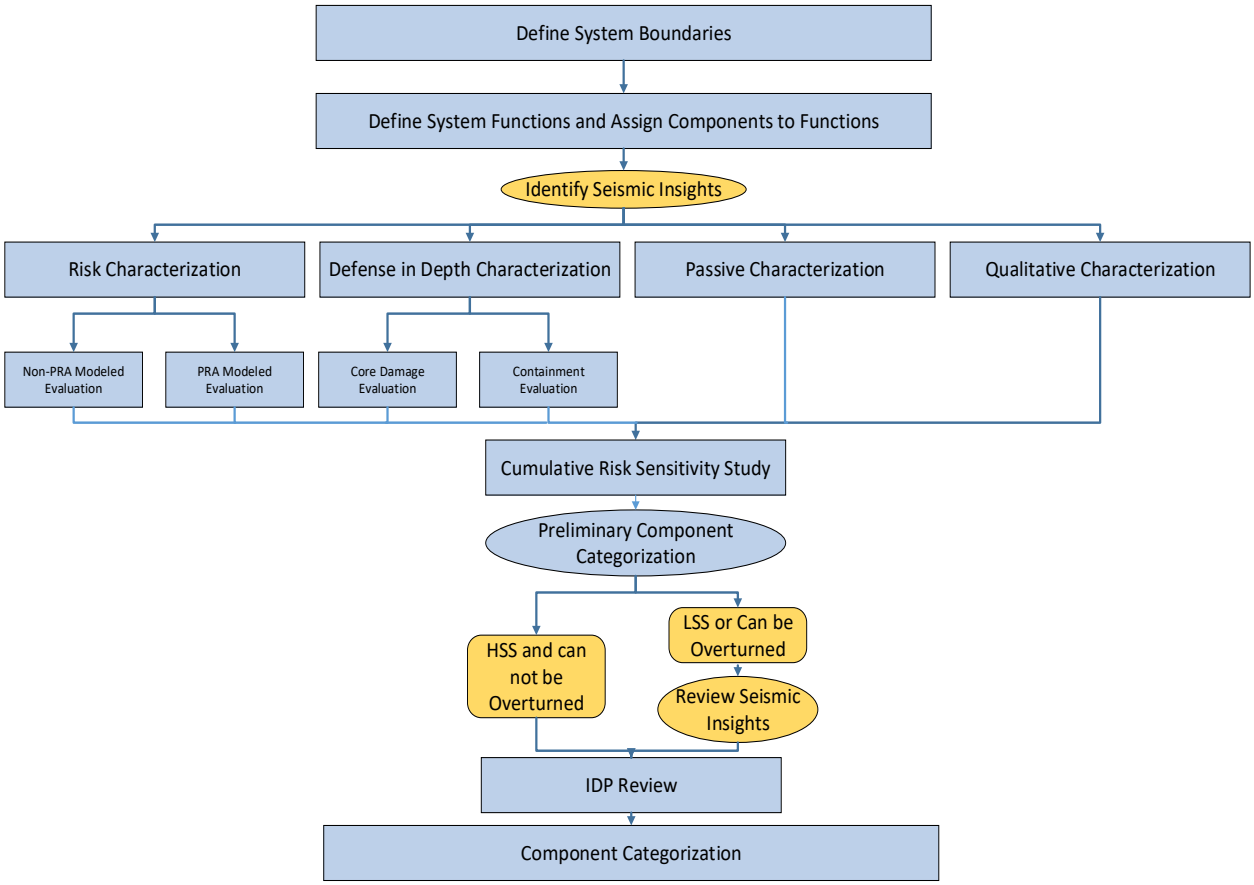
The process to categorize each system will be consistent with the guidance in NEI 00-04, "10 CFR 50.69 SSC Categorization Guideline," as endorsed by RG 1.201, with the exception of the evaluation of impact of the seismic hazard, which will use the EPRI 3002017583 (Reference [3]) approach for seismic Tier 2 sites, which includes JAF, 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)(1)(iv)." However, neither RG 1.201 nor NEI 00-04 prescribe a particular sequence or order for each of the elements to be completed. Therefore, the order in which each of the elements of the categorization process (listed below) is completed is flexible and as long as they are all completed, they may even be performed in parallel. Note that NEI 00-04 only requires Item 3 to be completed for components/functions categorized as 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 (SSEL), seismic safe shutdown equipment list (SSEL), other external events screening, and shutdown assessment)
3. Seven qualitative criteria in Section 9.2 of NEI 00-04
4. the defense-in-depth assessment
5. the passive categorization methodology

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 Low Safety Significant (LSS) that is presented to the Integrated Decision-Making Panel (IDP)). Note: the term "preliminary HSS or LSS" is synonymous with the NEI 00-04 term "candidate HSS or LSS." A component or function is preliminarily categorized as HSS if any element of the process results in a preliminary HSS determination in accordance with Table 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 Risk-Informed Safety Class (RISC) category can be assigned.

The IDP may direct and approve detailed categorization of components in accordance with NEI 00-04 Section 10.2. The IDP may always elect to change a preliminary LSS component or function to HSS, however the ability to change component categorization from preliminary HSS to LSS is limited. This ability is only available to the IDP for select process steps as described in NEI 00-04 and endorsed by RG 1.201. Table 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[1]	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[2]	N/A
Passive	Passive – Section 4	Segment/Component	Not Allowed	No

Notes:
[1]IDP consideration of seismic insights can also result in an LSS to HSS determination.

[2] 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.

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

an 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 JAF is a seismic Tier 2 (moderate 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 Integrated Decision-Making Panel (IDP) will be composed of a group of at least five experts who collectively have expertise in plant operation, design (mechanical and electrical) engineering, system engineering, safety analysis, and probabilistic risk assessment. At least three members of the IDP will have a minimum of five years of experience at the plant, and there will be at least one member of the IDP who has a minimum of three years of experience in the modeling and updating of the plant-specific PRA.

- The IDP will be trained in the specific technical aspects and requirements related to the categorization process. Training will address at a minimum the purpose of the categorization; present treatment requirements for SSCs including requirements for design basis events; PRA fundamentals; details of the plant specific PRA including the modeling, scope, and assumptions, the interpretation of risk importance measures, and the role of sensitivity studies and the change-in-risk evaluations; and the defense-in-depth philosophy and requirements to maintain this philosophy.

- The decision criteria for the IDP for categorizing SSCs as safety significant or low safety-significant pursuant to § 50.69(f)(1) will be documented in Exelon 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 High Safety Significant (HSS) and Low Safety Significant (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 sensitivity studies described in Section 8 of NEI 00-04. The factor of 3 was chosen as it is representative of the typical error factor of basic events used in the PRA model.

- 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 SER (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/Abnormal Operating Procedures, Exelon will not take credit for alternate means unless the alternate means are proceduralized and included in Licensed Operator training.

- JAF 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 2 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, as submitted to the NRC for TSTF 505 dated July 30, 2021, (ML21211A053) (Refer to Attachment 2).

- Fire Risks: Fire PRA model, as submitted to the NRC for TSTF 505 dated July 30, 2021, (ML21211A053) (Refer to Attachment 2).

- Seismic Risks: EPRI Alternative Approach in EPRI 3002017583 for Tier 2 plants with the markups provided in Attachment 2 of References [5] and [6] and additional considerations discussed in Section 3.2.3 of this license amendment request (LAR).

- Other External Risks (e.g., tornados, external floods): Using the IPEEE screening process as approved by NRC SE dated September 21, 2000, (TAC No. M83622) (Reference [7]). The other external hazards were determined to be insignificant contributors to plant risk.

- Low Power and Shutdown Risks: Qualitative defense-in-depth (DID) shutdown model for shutdown configuration risk management (CRM) based on the framework for DID provided in NUMARC 91-06, "Guidance for Industry Actions to Assess Shutdown Management"

(Reference [8]), 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 [9] (ML090930246) 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, if categorized, 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 Safety Evaluation 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 as published in Regulatory Guide 1.147, Revision 15. Both code cases employ a similar risk-informed safety classification of SSCs in order to change the repair/ replacement requirements of the affected LSS components. All ASME Code Class 1 SSCs with a pressure retaining function, as well as supports, will be 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 JAF 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 PRA models described below have been peer reviewed and there are no PRA upgrades that have not been peer reviewed. The PRA models described within this license amendment request (LAR) are the same as those described within the Exelon submittal of the LAR dated July 30, 2021, "License Amendment Request to Revise Technical Specifications to Adopt Risk Informed Completion Times TSTF-505, Revision 2, 'Provide Risk-Informed Extended Completion Times – RITSTF Initiative 4b,'" (ML21211A053) with routine maintenance updates applied.

3.2.1 Internal Events and Internal Flooding

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

3.2.2 Fire Hazards

The JAF categorization process for fire hazards will use a peer reviewed plant-specific fire PRA model. The internal Fire PRA model was developed consistent with NUREG/CR-6850 and only utilizes methods previously accepted by the NRC. The Exelon risk management process ensures that the PRA model used in this application reflects the as-built and as-operated plant for JAF. Attachment 2 at the end of this enclosure identifies the applicable Fire PRA model.

3.2.3 Seismic Hazards

10 CFR 50.69(c)(1) 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 (Reference [1]) 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 JAF seismic hazard assessment, Exelon 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 specified in Reference [3][1] with the EPRI markups provided in Attachment 2 of References [5] and [6] and includes additional considerations that are discussed in this section.

> (Note: The discussion below pertaining to Reference [3] includes the markups provided in Attachment 2 of References [5] and [6]).

The proposed categorization approach for JAF is a risk-informed graded approach that is demonstrated to produce categorization insights equivalent to a seismic PRA. This approach relies on the insights gained from the seismic PRAs examined in Reference [3] and plant specific insights considering seismic correlation effects and seismic interactions. Following the criteria in Reference [3], the JAF site is considered a Tier 2 site because the site Ground Motion Response Spectrum (GMRS) to SSE [Safe Shutdown Earthquake] comparison is above the Tier 1 threshold but not high enough that the NRC required the plant to perform an SPRA

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[1] EPRI 3002017583 is an update to EPRI 3002012988, "Alternative Approaches for Addressing Seismic Risk in 10 CFR 50.69 Risk-Informed Categorization," July 2018 (Reference [74]) which was referenced in the NRC issued amendment and SE for Calvert Cliffs Nuclear Power Plant, Units 1 and 2, to implement 10 CFR 50.69 as noted below:

(1) Calvert Cliffs Nuclear Power Plant, Units 1 and 2, "Issuance of Amendment Nos. 332 and 310 Re: Risk-Informed Categorization and Treatment of Systems, Structures, and Components (EPID L-2018-LLA-0482)," February 28, 2020. (ADAMS Accession No. ML19330D909) (Reference [75]).

(2) This license amendment request incorporates by Reference the Clinton Power Station, Unit 1 response to request for additional information letter of November 24, 2020 (ML20329A433) (Reference [76]), in particular, the response to the question regarding the differences between the initial EPRI report 3002012988 and the current EPRI report 3002017583 as well as Exelon's proposed approach for the 50.69 Seismic Alternative Tier 1.

to respond to Recommendation 2.1 of the Near Term Task Force 50.54(f) letter (Reference [10]). Reference [3] also demonstrates that seismic risk is adequately addressed for Tier 2 sites by the results of additional qualitative assessments discussed in this section and existing elements of the §50.69 categorization process specified in NEI 00-04.

The trial studies in Reference [3], as amended by their RAI responses and amendments (References [11], [12], [13], [14], [15], [16], [17]), [18], and [19]) 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 2 classification and resulting criteria 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 Reference [3].

> *"At Tier 2 sites, there may be a limited number of unique seismic insights, most likely attributed to the possibility of seismically correlated failures, appropriate for consideration in determining HSS SSCs. The special seismic risk evaluation process recommended using a Common Cause impact approach in the FPIE PRA can identify the appropriate seismic insights to be considered with the other categorization insights by the Integrated Decision-making Panel for the final HSS determinations."*

At sites with moderate seismic demands (i.e., Tier 2 range) such as JAF, there is no need to perform more detailed evaluations to demonstrate the inherent seismic capacities documented in industry sources such as Reference [20]. Tier 2 seismic demand sites have a lower likelihood of seismically induced failures and less challenges to plant systems than trial study plants. This, therefore, provides the technical basis for allowing use of a graded approach for addressing seismic hazards at JAF.

Test cases described in Section 3 of Reference [3], as amended by their RAI responses and amendments (References [11], [12], [13], [14], [15], [16], [17], [18], and [19]), showed that there are very few, if any, SSCs that would be designated HSS for seismic unique reasons. The test cases identified that the unique seismic insights were typically associated with seismically correlated failures and led to unique HSS SSCs. While it would be unusual even for moderate hazard plants to exhibit any unique seismic insights, it is prudent and recommended by Reference [3] to perform additional evaluations to identify the conditions where correlated failures and seismic interactions may occur and determine their impact in the 10 CFR 50.69 categorization process. The special sensitivity study recommended in Reference [3] uses common cause failures, similar to the approach taken in a FPIE PRA and can identify the appropriate seismic insights to be considered with the other categorization insights by the IDP for the final HSS determinations.

Exelon is using test case information from Reference [3], developed by other licensees. The test case information is being incorporated by Reference into this application, specifically Case

Study A (Reference [21]), Case Study C (Reference [22]), and Case Study D (Reference [23]) as well as, RAI responses and amendments (References [11], [12], [13], [14], [15], [16], [17], [18] and [19]), clarifying aspects these case studies.

Basis for JAF being a Tier 2 Plant

As defined in Reference [3], JAF meets the Tier 2 criteria for a "Moderate Seismic Hazard / Moderate Seismic Margin" site. The Tier 2 criteria are as follows:

> *"Tier 2: Plants where the GMRS [Ground Motion Response Spectrum] to SSE [Safe Shutdown Earthquake] comparison between 1.0 Hz and 10 Hz is greater than in Tier 1 but not high enough to be treated as Tier 3. At these sites, the unique seismic categorization insights are expected to be limited."*
>
> Note: Reference [3] applies to the Tier 2 sites in its entirety except for Sections 2.2 (Tier 1 sites) and 2.4 (Tier 3 sites).

For comparison, Tier 1 plants are defined as having a GMRS peak acceleration at or below approximately 0.2g or where the GMRS is below or approximately equal to the SSE between 1.0 Hz and 10 Hz. Tier 3 plants are defined where the GMRS to SSE comparison between 1.0 Hz and 10 Hz is high enough that the NRC required the plant to perform an SPRA to respond to the Fukushima 50.54(f) letter (Reference [10]).

As shown in Figure A4-1, comparing the JAF GMRS (derived from the seismic hazard) to the SSE (i.e., seismic design basis capability), the GMRS is below the SSE up to approximately 6 Hz and exceeds the SSE above 6 Hz and then drops back below at approximately 50 Hz (Reference [24]). The NRC screened out JAF from performing an SPRA in response to the NTTF 2.1 50.54(f) letter (Reference [25]). As such, it is appropriate that JAF is considered a Tier 2 plant. The basis for JAF being Tier 2 will be documented and presented to the IDP for each system categorized.

The following paragraphs describe additional background and the process to be utilized for the graded approach to categorize the seismic hazard for a Tier 2 plant.

Implementation of the Recommended Process

Reference [3] recommends a risk-informed graded approach for addressing the seismic hazard in the 10 CFR 50.69 categorization process. There are a number of seismic fragility fundamental concepts that support a graded approach and there are 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 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 (Reference [20]) 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.

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.

In applying the Reference [3] process for Tier 2 sites to the JAF 10 CFR 50.69 categorization process, the IDP will be provided with the rationale for applying the Reference [3] guidance and informed of plant SSC-specific seismic insights that the IDP may choose to consider in their HSS/LSS deliberations. 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 that provides identified plant seismic insights as well as the basis for applicability of the Reference [3] study and the bases for JAF being a Tier 2 plant. The discussion of the Tier 2 bases will include such factors as:

- The moderate seismic hazard for the plant,
- The definition of Tier 2 in the EPRI study, and
- The basis for concluding JAF is a Tier 2 plant.

At several steps of the categorization process, (e.g., as noted in Figure 3-1 and Table 3-1) 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 JAF) 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. For HSS

SSCs uniquely identified by the JAF PRA models but having design-basis functions during seismic events or functions credited for mitigation and prevention of severe accidents caused by seismic events, these will be addressed using non-PRA based qualitative assessments in conjunction with any seismic insights provided by the PRA.

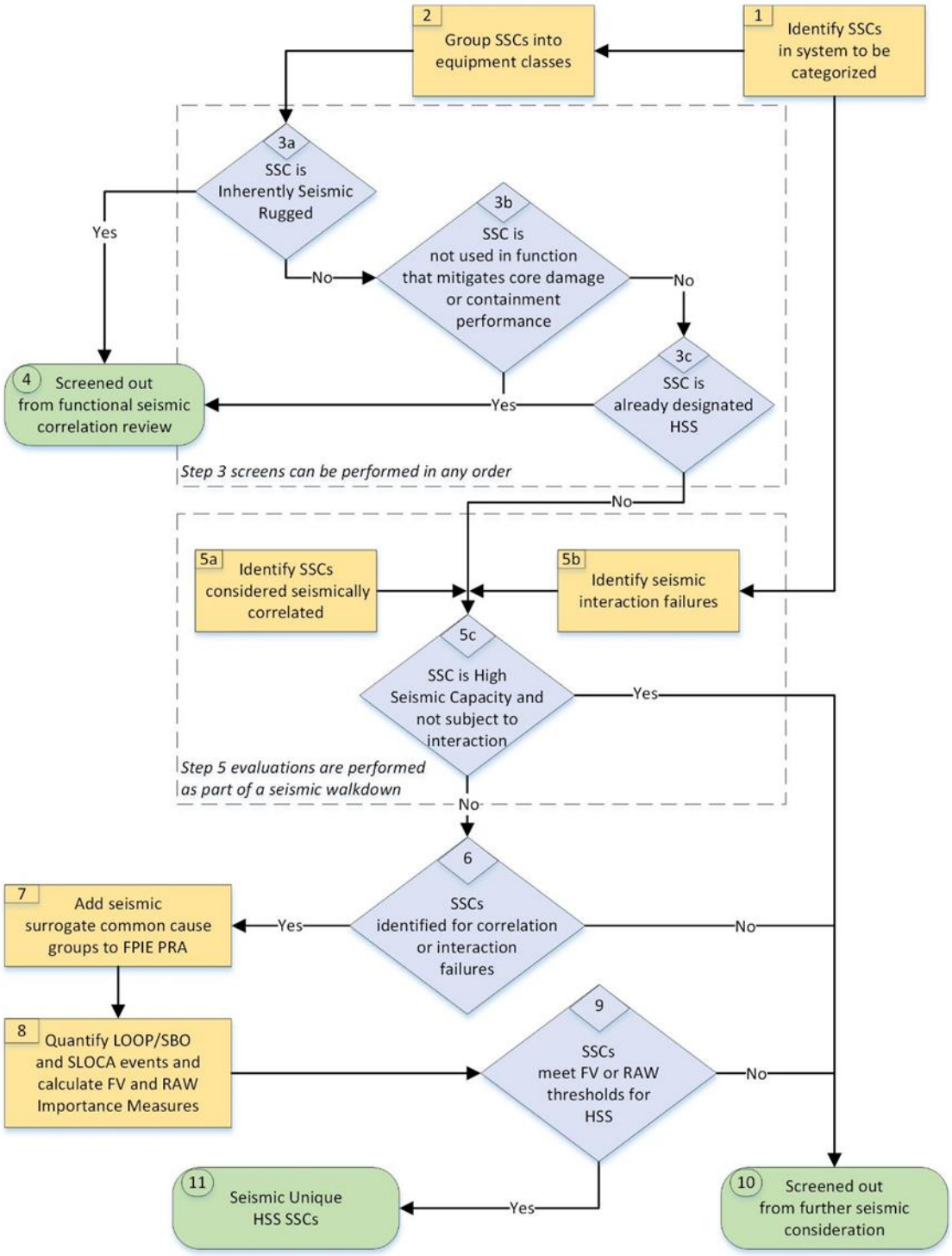
For components that are HSS due to fire PRA but not HSS due to internal events PRA, the categorization team will review design-basis functions during seismic events or functions credited for mitigation and prevention of severe accidents caused by seismic events and characterize these for presentation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

The categorization team will review available JAF plant-specific seismic reviews and other resources such as those identified above. The objective of the seismic review is to identify plant-specific seismic insights 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 implicitly part of PRA-modeled functions (including relays)

For each system categorized, the categorization team will evaluate correlated seismic failures and seismic interactions between SSCs. This process is detailed in Reference [3] Section 2.3.1 and is summarized below in Figure 3-2.

Figure 3-2: Seismic Correlated Failure Assessment for Tier 2 Plants[2]



[2] Reproduced from Reference [3] Figure 2-3 including the markups provided in Attachment 2 of References [5] and [6].

Determination of seismic insights will make use of the full power internal events PRA model supplemented by focused seismic walkdowns. An overview of the process to determine the importance of SSCs for mitigating seismic events follows and is utilized on a system basis:

- Identify SSCs within the system to be categorized
- Group SSCs within the system into the classes of equipment and distributed systems used for SPRAs.
- Refine the list and screen out the following SSCs from consideration of functional correlated seismic failures:
    - Inherently rugged components
    - Components not used in safety functions that support mitigation of core damage or containment performance
    - Components already identified as HSS components from the Internal Events PRA or Integrated assessment
- Perform a seismic walkdown:
    - For SSCs screened IN look for correlation
    - For SSCs screened IN or OUT assess for spatial interaction concerns that could fail multiple components in the system, or could fail a single component in the system due to either seismic interaction or direct component failure modes, that result in total loss of a multi-train system and where there is not another system that independently provides the same function
- Based on the seismic walkdown:
    - Screen out IF SSCs have high seismic capacity AND not included in seismically correlated groups or correlated interaction groups
- Add surrogate events to the FPIE model that simulate spatial interaction or Correlation (for the system being categorized) - set the probability of failure to 1E-04.
- Quantify the FPIE model (for the system being categorized) for LOOP and Small LOCA (SLOCA) initiated accident sequences setting (1) the LOOP initiating event frequency to 1.0/yr, (2) the SLOCA initiating event frequency to 1E-02/yr, and (3) the initiating event frequency for all initiators other than LOOP and small LOCA initiators to 0 (zero), and also removing credit for restoration of offsite power in LOOP/SBO accident sequences as well as other functional recoveries
- Utilize the Importance Measures from this sensitivity study to identify appropriate SSCs (in the system being categorized) that should be HSS due to correlation or seismic interactions

Seismic impacts would be compiled on an SSC basis. As each system is categorized, the system-specific seismic insights will be documented in the categorization report and provided to the IDP for consideration as part of the IDP review process (e.g., Figure 3-1). The IDP cannot challenge any candidate HSS recommendation for any SSC from a seismic perspective if they believe there is a basis, except for certain conditions identified in Step 10 of Section 2.3.1 of Reference [3]. Any decision by the IDP to downgrade preliminary HSS components to LSS will consider the applicable seismic insights in that decision. SSCs identified from the Fire PRA as candidate HSS, which are not HSS from the internal events PRA or integrated importance measure assessment, will be reviewed for their design basis function during seismic events or functions credited for mitigation and prevention of severe accidents caused by seismic events. These insights will provide the IDP a means to consider potential impacts of seismic events in the categorization process.

If the JAF seismic hazard changes from medium risk (i.e., Tier 2) at some future time, prior NRC approval, under 10 CFR 50.90, will be requested if JAF's feedback process determines that a process different from the proposed alternative seismic approach is warranted for seismic risk consideration in categorization under 10 CFR 50.69. After receiving NRC approval, Exelon will follow its categorization review and adjustment process to review the changes to the plant and update, as appropriate, the SSC categorization in accordance with 10 CFR 50.69(e) and the EPRI 3002017583 SSC categorization criteria for the updated Tier. This includes use of the Exelon corrective action process (CAP).

If the seismic hazard is reduced such that it meets the criteria for Tier 1 in EPRI 3002017583, Exelon will implement the following process.

a) For previously completed system categorizations, Exelon may review the categorization results to determine if use of the criteria in EPRI 3002017583 Section 2.2, "Low Seismic Hazard / High Seismic Margin Sites" would lead to categorization changes. If changes are warranted, they will be implemented through the Exelon design control and corrective action programs and NEI 00-04, Section 12.

b) Seismic considerations for subsequent system categorization activities will be performed in accordance with the guidance in EPRI 3002017583 Section 2.2, "Low Seismic Hazard / High Seismic Margin Sites."

If the seismic hazard increases to the degree that a seismic probabilistic risk assessment (SPRA) becomes necessary to demonstrate adequate seismic safety, Exelon will implement the following process following completion of the SPRA, including adequate closure of Peer Review Findings and Observations.

a) For previously completed system categorizations, Exelon will review the categorization results using the SPRA insights as prescribed in NEI 00-04 Section 5.3, Seismic Assessment and Section 5.6, "Integral Assessment". If changes are warranted, they will be implemented through the Exelon design control and corrective action programs and NEI 00-04 Section 12.

b) Seismic considerations for subsequent system categorization activities will follow the guidance in NEI 00-04, as recommended in EPRI 3002017583 Section 2.4, "High Seismic Hazard / Low Seismic Margin Sites".

Historical Seismic References for JAF

The JAF GMRS and SSE curves from the seismic hazard and screening response are shown in Section 2.4 and 3.1, respectively, in the seismic hazard and screening report of Reference [26]. The JAF Safe Shutdown Earthquake (SSE) and GMRS curves from Reference [26] are shown in Figure A4-1 in Attachment 4. The NRC's Staff assessment of the JAF seismic hazard and screening response is documented in Reference [25]. In the Staff Confirmatory Analysis (Section 3.3.3) of Reference [25], the NRC concluded that the methodology used by Exelon in determining the GMRS was acceptable and that the GMRS determined by Exelon adequately characterizes the reevaluated hazard for the JAF 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 JAF, 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 (References [26] and [25]).

2. NTTF Recommendation 2.1 spent fuel pool assessment (References [27] and [28]).

3. NTTF Recommendation 2.3 seismic walkdowns (References [29] and [30]).

4. NTTF Recommendation 4.2 seismic mitigation strategy assessment (S-MSA) (References [31] and [32]).

The following additional post-Fukushima seismic reviews were performed for JAF:

5. NTTF Recommendation 2.1 Expedited Seismic Evaluation Process (ESEP) (References [33] and [34]).

6. NTTF Recommendation 2.1 seismic High Frequency Evaluation (References [35] and [36].

Summary

Based on the above, the Summary from Section 2.3.3 of Reference [3] applies to JAF; namely, JAF is a Tier 2 plant for which there may be a limited number of unique seismic insights, most likely attributed to the possibility of seismically correlated failures, appropriate for consideration in determining HSS SSCs. References [5], [6], and [37][3] are incorporated into this LAR as they provide additional supporting bases for Tier 2 plants. In addition, References [38], [39], and [40] are incorporated into this LAR as they provide additional supporting bases for Tier 1 plants that is also used for Tier 2 plants. The special sensitivity study recommended using common cause failures, similar to the approach taken in a FPIE PRA, can identify the appropriate seismic insights to be considered with the other categorization insights by the Integrated Decision-making Panel (IDP) for the final HSS determinations. Use of the EPRI approach outlined in Reference [3] to assess seismic hazard risk for §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 §50.69(c).

3.2.4 Other External Hazards

All external hazards, except for seismic, were screened for applicability to JAF per a plant-specific evaluation in accordance with GL 88-20 (Reference [41]) and updated to use the criteria in ASME PRA Standard RA-Sa-2009. Attachment 4 provides a summary of the external hazards screening results. Attachment 5 provides a summary of the progressive screening approach for external hazards.

3.2.5 Low Power & Shutdown

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

[3] Excludes RAI APLC 50.69-RAI No. 12 that addresses a non-seismic topic (external events).

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 NEI 00-04 will be considered preliminary HSS.

3.2.6 PRA Maintenance and Updates

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

3.2.7 PRA Uncertainty Evaluations

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

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

In the overall risk sensitivity studies, Exelon will utilize a factor of 3 to increase the unavailability or unreliability of LSS components consistent with that approved for Vogtle in Reference [4]. Consistent with the NEI 00-04 guidance, Exelon will perform both an initial sensitivity study and

a cumulative sensitivity study. The initial sensitivity study applies to the system that is being categorized. In the cumulative sensitivity study, the failure probabilities (unreliability and unavailability, as appropriate) of all LSS components modeled in all identified PRA models for all systems that have been categorized are increased by a factor of 3. This sensitivity study together with the periodic review process assures that the potential cumulative risk increase from the categorization is maintained acceptably low. The performance monitoring process monitors the component performance to ensure that potential increases in failure rates of categorized components are detected and addressed before reaching the rate assumed in the sensitivity study.

The detailed process of identifying, characterizing, and qualitative screening of model uncertainties is found in Section 5.3 of NUREG-1855 and Section 3.1.1 of EPRI TR-1016737 (Reference [42]). 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.

Each PRA element notebook was reviewed for assumptions and sources of uncertainties. The characterization of assumptions and sources of uncertainties are based on whether the assumption and/or source of uncertainty is key to the 10 CFR 50.69 application in accordance with RG 1.200 Revision 2.

Key JAF 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 JAF PRA model specific assumptions or sources of uncertainty.

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

The PRA models described in Section 3.2 have been assessed against RG 1.200, "An Approach for Determining the Technical Adequacy of Probabilistic Risk Assessment Results for Risk-Informed Activities," Revision 2 (Reference [43]), consistent with NRC RIS 2007-06.

Finding and Observation (F&O) closure reviews were conducted on the PRA models discussed in this section. Closed Findings were reviewed and closed using the process documented in Appendix X to NEI 05-04, NEI 07-12 and NEI 12-13, “Close-out of Facts and Observations” (F&Os) (Reference [44]] as accepted by NRC in the letter dated May 3, 2017 (Reference [45]). The results of this review have been documented and are available for NRC audit.

<u>Full Power Internal Events and Internal Flooding PRA Model</u>

The JAF FPIE PRA model was peer reviewed in September 2009 using the NEI 05-04 process (Reference [46]). the PRA Standard (ASME/ANS RA-Sa-2009) (Reference [47], and Regulatory Guide 1.200, Revision 2 (Reference [43]). This Peer Review (Reference [48]) was a full-scope

review of the technical elements of the Internal Events and Internal Flooding, at-power PRA. The Findings from the Peer Review have been addressed in the Internal Events PRA model.

In November 2019, an F&O Closure Review (Reference [49]) was conducted for JAF. The scope of the review included the Internal Events and Internal Flooding PRA model. The F&O Independent Assessment Team closed twenty-three of twenty-four Finding level F&Os and the re-assessments for linked SRs were deferred to a concurrent focused-scope Peer Review. The concurrent focused-scope Peer Review (Reference [50]) was performed in November 2019 and the review team determined there were six Finding level F&Os resulting in three Not-Met SRs. An additional F&O Closure by Independent Assessment (Reference [51]) was held in July 2020 to review close out of Finding level F&Os from the three prior PRA Peer Reviews against the PRA Standard. All seven Finding level F&Os from the FPIE PRA were assessed as closed during the review. Currently, there are no open Finding level F&Os against the FPIE PRA model.

Fire PRA Model

The JAF Fire PRA (FPRA) Peer Review (Reference [52] was performed in January 2020 using the NEI 07-12 Fire PRA peer review process (Reference [53]), the ASME PRA Standard, ASME/ANS RA-Sa-2009 (Reference [47]), and Regulatory Guide 1.200, Rev. 2 (Reference [43]). The FPRA Peer Review was a full-scope review of all technical elements of the JAF at-power FPRA against all technical elements in Part 4 of the ASME/ANS PRA Standard, including the referenced Internal Events Supporting Requirements (SRs) in Part 2.

The Fire PRA Peer Review team determined there were thirty-nine Finding level F&Os resulting in twenty-three Not-Met SRs. In July 2020, an F&O Closure by Independent Assessment (Reference [51]) was conducted for JAF. The scope of the review included FPIE and Fire PRA Peer Review Finding level F&Os. Thirty-three of the FPRA Finding level F&Os were assessed as closed during the review. In March 2021, a follow-on F&O Closure by Independent Assessment (Reference [54]) was conducted and the remaining open Findings were assessed as closed. Currently, there are no open Finding level F&Os against the FPRA model.

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

The JAF 10 CFR 50.69 categorization process will implement the guidance in NEI 00-04. The overall risk evaluation process described in the NEI guidance addresses both known degradation mechanisms and common cause interactions and meets the requirements of §50.69(b)(2)(iv). Sensitivity studies described in NEI 00-04 Section 8 will be used to confirm that the categorization process results in acceptably small increases to core damage frequency (CDF) and large early release frequency (LERF). The failure rates for equipment and initiating event frequencies used in the PRA include the quantifiable impacts from known degradation mechanisms, as well as other mechanisms (e.g., design errors, manufacturing deficiencies, 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 JAF Tier 2 approach discussed in section 3.2.3, implementation of the Exelon 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 is described in Exelon's 10 CFR 50.69 program documents. The program requires that the periodic review assess changes that could impact the categorization results and provides the Integrated Decision-making Panel (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 help catch and reverse negative performance trends and take corrective action if necessary.

The Exelon 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 has been updated to 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.

Exelon has a 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 Exelon 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 no longer than once every two refueling outages 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 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 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

Exelon 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.

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

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

Response: No.

The proposed change will permit the use of a risk-informed categorization process to modify the scope of 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, Exelon concludes that the proposed change presents no significant hazards consideration under the standards set forth in 10 CFR 50.92(c), and, accordingly, a finding of "no significant hazards consideration" is justified.

4.3 CONCLUSIONS

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

5 ENVIRONMENTAL CONSIDERATION

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

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Attachment 1: List of Categorization Prerequisites

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

- Integrated Decision-Making Panel (IDP) member qualification requirements
- Qualitative assessment of system functions. System functions are qualitatively categorized as preliminary 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). 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.
- 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.

Attachment 2: Description of PRA Models Used in Categorization

Unit	Model	Baseline CDF	Baseline LERF	Comments
Full Power Internal Events (FPIE) and Internal Flooding PRA Model				
1	JF117A Peer Reviewed Against RG 1.200 R2 in September 2009	3.2E-06	5.7E-07	2020 FPIE Application Specific Model
Fire (FPRA) Model				
1	JF2017CF4 Peer Reviewed Against RG 1.200 R2 in January 2020	1.9E-05	3.9E-06	2021 Fire PRA Application Specific Model

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

Finding Number	Supporting Requirement(s)	Capability Category (CC)	Description	Disposition for 10 CFR 50.69
There are no open peer review findings or self-assessment open items				

Attachment 4: External Hazards Screening[4]

Hazard	Definition	Screening Criteria	Disposition for 10 CFR 50.69
Aircraft Impact	An aircraft (either a portion of (e.g., missile) or the entire aircraft) that collides either directly or indirectly (i.e., skidding impact with one or more structures, systems, or components (SSCs) at or in the plant's analyzed area causing functional failure. Secondary hazards resulting from an aircraft impact include, but are not necessarily limited to, fire.	PS2 PS4 C3	Acceptance criterion 1.A of Standard Review Plan 3.5.1.6 (Reference [55]) states the probability is considered to be less than an order of magnitude of 10^{-7} per year by inspection if the plant-to-airport distance D is between 5 and 10 statute miles, and the projected annual number of operations is less than 500 D^2, or the plant-to-airport distance D is greater than 10 statute miles, and the projected annual number of operations is less than 1000 D^2 **(PS2, PS4)**. The closest airport to the plant is the Oswego County Airport, a small, public, general aviation facility located approximately 11 miles south of the plant. According to the Federal Aviation Administration's Air Traffic Activity System (Reference [56]), the annual operations from this airport is less than 21,000, which is less than the 500 D^2 criteria **(PS2, PS4)**. Syracuse International Airport, about 30 miles southwest of the plant, is the nearest airport with scheduled commercial air service. According to the Federal Aviation

[4] The list of hazards and their potential impacts considered those items listed in Tables D-1 and D-2 in Appendix D of RG 1.200, Rev. 3 (Reference [77]).

Hazard	Definition	Screening Criteria	Disposition for 10 CFR 50.69
			Administration's Air Traffic Activity System (Reference [56]), the annual operations from this airport is less than ~70,000, which is less than the 1000 D^2 criteria **(PS2, PS4)**. Per IPEEE 5.5.1.3 (Reference [57]), JAF is more than 5 statute miles from the edge of the nearest military training route (**C3**). Based on this review, the aircraft impact hazard is considered to be negligible.
Avalanche	Rapid flow of a large mass of accumulated frozen precipitation and other debris down a sloped surface resulting in dynamic loading of SSCs at or in the plant's analyzed area causing functional failure or adverse impact on natural water supplies used for heat rejection.	C3	JAF is located on the southeast shore of Lake Ontario, which precludes the possibility of an avalanche. Based on this review, the Avalanche hazard can be considered to be negligible.
Biological Events	Accumulation or deposition of vegetation or organisms (e.g., zebra mussels, clams, fish, algae) on an intake structure or internal to a system that uses raw cooling water from a source of surface water,	C5	This hazard is slow to develop and can be identified via monitoring and managed via standard maintenance process. Actions committed to and completed by JAF in response to Generic Letter 89-13 provide on-going control of biological hazards. These controls are described in Exelon procedure ER-AA-340, "GL 89-13 Program

Hazard	Definition	Screening Criteria	Disposition for 10 CFR 50.69
	causing its functional failure.		Implementing Procedure (Reference [58]). Additional actions to reduce potential intake cooling water blockages were performed in response to WANO SOER 07-2 (Reference [59]). Based on this review, the Biological Event hazard can be considered to be negligible.
Coastal Erosion	Removal of material from a shoreline of a body of water (e.g., river, lake, ocean) due to surface processes (e.g., wave action, tidal currents, wave currents, drainage, or winds and including river bed scouring) that results in damage to the foundation of SSCs at or in the plant's analyzed area, causing functional failure.	C1 C5	The lake water Intake Structure is a reinforced concrete structure setting on the lake bottom at a distance of approximately 900 feet from the shoreline in approximately 25 feet of water. The main structure is anchored to the natural bedrock below the lake bottom by post-tensioned tendons (Reference [59]). No major structures are directly on the coastline, so coastal erosion is not a significant concern at JAF. JAF has not witnessed a change in the coastline or in the lakebed levels in the area of the intake structure (Reference [59]) **(C1)**. Additionally, coastal erosion would slowly develop, so any effects could be mitigated **(C5)**. Based on this review, the Coastal Erosion hazard can be considered to be negligible.
Drought	A shortage of surface water supplies due to a period of below-average	C5	Drought is a slowly developing hazard allowing time for orderly

Hazard	Definition	Screening Criteria	Disposition for 10 CFR 50.69
	precipitation in a given region, thereby depleting the water supply needed for the various water-cooling functions at the facility.		plant reductions, including shutdowns. Based on this review, the Drought hazard can be considered to be negligible.
External Flood	An excess of water outside the plant boundary that causes functional failure to plant SSCs. External flood causes include, but may not be limited to, flooding due to dam failure, high tide, hurricane (tropical cyclone), ice cover, local intense precipitation, river diversion, river and stream overflow, seiche, storm surge, and tsunami.	C1	The FHRR identified both LIP stillwater and PMF maximum water surface elevation (WSE) of 272.8 ft. This is greater than the existing CDB controlling flood elevation of 262 ft and is slightly above site grade, which is nominally 272 ft. In-leakage from exterior doors that are normally closed was evaluated in JAF-RPT-14-00035 (Reference [60]) to quantify the volume of water that could possibly enter any buildings from the 0.8 ft of ponding during either of these flood events. The conclusions were that in-leakage will be minimal and interior drainage features would have more than enough capacity to mitigate the effects of any in-leakage from the normally closed exterior doors. Procedure AOP-13 (Reference [61]) directs operators to verify water intrusion is not occurring at building outer doors and to close doors as appropriate if sustained local intense precipitation is occurring. Otherwise, no human actions are required to mitigate the effects of a flooding event at the station.

Hazard	Definition	Screening Criteria	Disposition for 10 CFR 50.69
			There are several doors whose failure to be in their normal position would result in an unscreened scenario. These doors are a mixture of personnel doors, roll up doors, and vertical hatches that are required to be closed. Therefore, these doors and barriers should be categorized as HSS in accordance with NRC-approved guidance since removal of any of these doors would result in an unscreened scenario. With credit taken for the normally positioned personnel doors, external flooding mechanisms are screened utilizing Criterion C1 and will not impact 10 CFR 50.69 categorization of other SSCs.
Extreme Winds and Tornadoes	Strong winds resulting in dynamic loading or missile impacts on SCCs causing functional failure. Hazards that could potentially result in high wind include the following: • hurricane - severe winds developed from a tropical depression resulting in missiles or dynamic loading on SSCs. Secondary	PS4	Based on the plant design for tornado wind pressure and the very low frequency (<1E-7/yr) of occurrence of design basis tornadoes, a demonstrably conservative estimate of CDF associated with high wind hazard (other than wind-generated missiles) is much less than 1E-6/yr. Therefore, all non-missile high wind hazards can be screened from consideration for the 10 CFR 50.69 application, based on Criterion C of Supporting Requirement EXT-C1 of ASME/ANS RA-Sa-2009 (Reference [47]).

Hazard	Definition	Screening Criteria	Disposition for 10 CFR 50.69
	hazards resulting from a hurricane, include, but are not necessarily limited to tornado • straight wind - a strong wind resulting in missiles or dynamic loading on SSCs that is not associated with either hurricanes or tornadoes • tornado - a strong whirlwind that results in missiles or dynamic loading on SSCs Fog Low-lying water vapor in the form of a cloud or obscuring haze of atmospheric dust or smoke resulting in impeded visibility that could result in, for example, a transportation accident.		Based on a plant-specific tornado missile risk analysis for JAF (Reference [62]), the CDF for tornado missiles is conservatively estimated to be less than 1E-6/yr. Therefore, tornado missile hazards can be screened from consideration for the 10 CFR 50.69 application, based on Criterion C of Supporting Requirement EXT-C1 of ASME/ANS RA-Sa-2009 (Reference [47]). There are no vulnerabilities to tornado missiles at JAF that would specifically affect containment integrity and large early release probability. There are no SSCs credited in the screening determination of high winds and tornado missile hazards, including passive and/or active components, other than Seismic Category I structures which are already considered high safety significant (HSS) for 10 CFR 50.69 categorization.
Fog	Low-lying water vapor in the form of a cloud or obscuring haze of atmospheric dust or smoke resulting in impeded visibility that could result in, for example, a transportation accident.	C4	The principal effects of such events (such as freezing fog) would be to cause a loss of off-site power, which is addressed in weather-related LOOP scenarios in the FPIE PRA model for JAF. Based on this review, the Fog hazard can be considered to be negligible.

Hazard	Definition	Screening Criteria	Disposition for 10 CFR 50.69
Forest Fire	Direct (e.g., thermal effects) and indirect effects (e.g., generation of combustion products, transport of firebrand) of a forest fire outside the plant boundary that causes functional failure of plant SSCs. Hazards that could cause or be caused by a forest fire include, but may not be limited to, wildfires and grass fires.	C3 C4	The UFSAR Section 2.1.1 refers to the "NMP-JAF" site since the Nine Mile Point (NMP) and JAF plants are essentially geographically co-located. (Reference [63]). The JAF IPEEE and UFSAR do not discuss this hazard in any great detail; however, the hazard is discussed in sufficient detail and was screened in the NMP2 10 CFR 50.69 application (Reference ADAMS Accession No. ML19360A145) and confirmed in the NMP2 NRC Safety Evaluation (ADAMS Accession No. ML20332A115). Per the NMP2 USAR Section 2.2.3.1.4 (Reference [64]), the site is sufficiently cleared in areas adjacent to the plant that forest or brush fires pose no safety hazards. (**C3**). In addition, external fires (Forest or Range Fires) originating from outside the plant boundary have the potential to cause a loss of offsite power event, which is addressed for grid-related LOOP scenarios in the FPIE PRA model for JAF (**C4**). Based on this review, the Forest or Range Fire hazard can be considered to be negligible.
Frost	A thin layer of ice crystals that form on the	C4	The principal effects of such events would be to cause a loss of

Hazard	Definition	Screening Criteria	Disposition for 10 CFR 50.69
	ground or the surface of an earthbound object when the temperature of the ground or surface of the object falls below freezing.		off-site power, which is addressed for weather-related LOOP scenarios in the FPIE PRA model for JAF. Based on this review, the Frost hazard can be considered to be negligible.
Hail	A shower of ice or hard snow that could result in transportation accidents or directly causes dynamic loading or freezing conditions as a result of ice coverage.	C4	The principal effects of such events would be to cause a loss of off-site power, which is addressed for weather-related LOOP scenarios in the FPIE PRA model for JAF. Based on this review, the Hail hazard can be considered to be negligible.
High Summer Temperature	Effects on SSC operation due to abnormally high ambient temperatures resulting from weather phenomena. Secondary hazards resulting from high ambient temperatures, include, but are not necessarily limited, to low lake or river water levels.	C1 C5 C4	Per UFSAR Section 7.1.12 (Reference [63]), the plant is designed for this hazard (**C1**). The principal effects of such events would result in elevated lake temperatures, which are monitored by station personnel. Actions would be taken in response to elevated lake temperatures (**C5**). In addition, plant trips due to this hazard are covered in the definition of another event in the PRA model (e.g., transients, loss of condenser) (**C4**).

Hazard	Definition	Screening Criteria	Disposition for 10 CFR 50.69
			Based on this review, the High Summer Temperature hazard can be considered to be negligible.
High Tide	The periodic maximum rise of sea level resulting from the combined effects of the tidal gravitational forces exerted by the Moon and Sun and the rotation of the Earth.	C1	The evaluation of the impact of the external flooding hazard at the site was updated as a result of the NRC's post Fukushima 50.54(f) Request for Information. The station's flood hazard reevaluation report (FHRR) was submitted to NRC for review (Reference [65]) and NRC issued an interim staff response letter (Reference [66]) confirming the findings in the FHRR. The results indicate that all flood causing mechanisms, except Local Intense Precipitation (LIP), flooding in streams and rivers (herein referred to as the PMF), and storm surge (herein referred to as Combined Effects flooding) are bounded by the current design basis (CDB) and do not pose a challenge to the plant. Therefore, high tide, lake level, and river stage hazard impacts are considered negligible and can be screened **(C1)**.
Hurricane (Tropical Cyclone)	Flooding that results from the intense rain fall from a hurricane (tropical cyclone). Secondary hazards resulting from a hurricane include, but are not necessarily	C3 C4	JAF is approximately 250 miles from the Atlantic Ocean. The risk associated with hurricane hazards do not need to be evaluated that far inland (**C3**); see Supporting Requirement WHA-A2, Note (2) in the ASME/ANS PRA Standard (Reference [47]).

Hazard	Definition	Screening Criteria	Disposition for 10 CFR 50.69
	limited to, dam failure, high tide, river and stream overflow, seiche, storm surge, and waves.		Wind and storm surge impacts on the plant due to hurricanes are bounded by the Extreme Wind / Tornado and External Flooding hazards (**C4**). Based on this review, the Hurricane hazard can be considered to be negligible.
Ice Cover	Flooding due to downstream blockages of ice on a river. Secondary hazards resulting from an ice blockage include, but are not necessarily limited to, river and stream overflow.	C1 C4	UFSAR Section 2.3.5 (Reference [63]) states that Lake Ontario rarely, if ever, freezes over completely. However, ice floes from the Niagara River and ball ice and slush formed on the lake are driven by wind and frequently pack against the shoreline. Plant SSCs are designed for this hazard. The lake water Intake Structure is a reinforced concrete structure setting on the lake bottom at a distance of approximately 900 ft from the shoreline in approximately 25 ft of water. The Screenhouse is approximately 150 ft from the shore; this location was chosen in lieu of the conventional shoreline intake because of the large masses of ice which build up along the south shore of Lake Ontario every year (Reference [59]). (**C1**) A potential impact of ice cover on other SSCs is a loss of off-site power event, which is addressed for weather-related LOOP scenarios in the FPIE PRA model for JAF (**C4**).

Hazard	Definition	Screening Criteria	Disposition for 10 CFR 50.69
			Based on this review, the Ice Cover hazard can be considered to be negligible.
Industrial or Military Facility Accident	An accident at an offsite industrial or military facility that results in a release of toxic gases, a release of combustion products, a release of radioactivity, an explosion, or the generation of missiles.	C3 PS4	As described in IPEEE 5.2.3 and 5.5.2.1, progressive screening steps were taken to assess hazardous chemical, transportation, and nearby industrial facility incidents (there are no military facilities within 5 miles of the site). No accident scenarios were calculated to core damage frequency (CDF) exceeding 10^{-6}/year (**C3**, **PS4**). A similar analysis was performed to identify potential explosion incidents that could give rise to a 1-psi overpressure at the plant. No scenario resulted in an overpressure greater than 1 psi or missile at the site, per Table 5.5.2.4 of the UFSAR (**C3**). Table 2.1-4 of the UFSAR contains an updated list of facilities; these were either addressed in the IPEEE or are too far from the site to have an impact. See also Toxic Gas. Based on this review, the Industrial or Military Facility Accident hazard can be considered to be negligible.

Hazard	Definition	Screening Criteria	Disposition for 10 CFR 50.69
Internal Flood	Flooding that results from leaks or ruptures of liquid systems (e.g., tanks, pipes, valves, pumps) originating inside the defined plant site boundary.	N/A	The JAF Internal Events PRA includes evaluation of risk from internal flooding events.
Landslide	Dynamic loading of SSCs or impacts on natural water supplies used for heat rejection due to the movement of rock, soil, and mud down a sloped surface (does not include frozen precipitation).	C3	Plant site is located on level terrain and is not subject to landslides. Based on this review, the Landslide hazard can be considered to be negligible.
Lightning	Effects on SSCs due to a sudden electrical discharge from a cloud to the ground or Earth-bound object.	C4	Lightning strikes are not uncommon in nuclear plant experience. They can result in losses of off-site power or surges in instrumentation output if grounding is not fully effective. The latter events often lead to reactor trips. Both events are incorporated into the JAF internal events model through the incorporation of generic and plant-specific data. Based on this review, the Lightning hazard can be considered to be negligible.
Low Lake or River Water Level	A decrease in the water level of the lake or river used for power generation.	C5	UFSAR 2.4.3.6 (Reference [63]) defines the design minimum low water level of Lake Ontario for JAF as el. 236.5 ft. This is based on a minimum still water level of 240.6’

Hazard	Definition	Screening Criteria	Disposition for 10 CFR 50.69
			and instantaneous lowering of 4.1 ft due to a maximum probable seiche on Lake Ontario. Historical low water levels are also provided; the lowest recorded since 1860 is el. 242.6 ft. The UFSAR provides a description of the regulation of Lake Ontario by the International Lake Ontario-St. Lawrence River Board. UFSAR Section 2.4.3.5 states the lake level is regulated within a range of approximately 242 to 249 ft. The basic water supply to the lake changes very slowly, permitting reasonably accurate forecasts and operating actions to maintain desired levels. Section 2.4.3.6 of the UFSAR also describes the effect of the failure of the St. Lawrence Power Project, which consists of two dams and a hydroelectric power plant in the St. Lawrence River (the outlet of Lake Ontario). In the unlikely event that all dams simultaneously fail, it is estimated that Lake Ontario water levels would decline gradually, with the full effect experienced about a year following the failure. In this time the still water level could fall to a minimum at el. 240.6 ft. Due to the normal regulation of lake level and the extended time available before minimum water

Hazard	Definition	Screening Criteria	Disposition for 10 CFR 50.69
			level in the event of dam failures downstream, this hazard can be screened based on Criterion C5. Based on this review, the Low Tide, Lake Level, or River Stage hazard can be considered to be negligible.
Low Winter Temperature	Effects on SSC operation due to abnormally low ambient temperatures resulting from weather phenomena. Secondary hazards resulting from low ambient temperatures include, but are not necessarily limited to, frost, ice cover, and snow.	C5 C4	The principal effects of such events would be to cause a loss of off-site power. These effects would take place slowly allowing time for orderly plant reductions, including shutdowns (**C5**). At worst, the loss of off-site power events would be subsumed into the base PRA model results (**C4**). Based on this review, the Low Winter Temperature hazard can be considered to be negligible.
Meteorite/Satellite Strikes	A release of energy due to the impact of a space object such as a meteoroid, comet, or human-caused satellite falling within the Earth's atmosphere, a direct impact with the Earth's surface, or a combination of these effects. This hazard is analyzed with respect to direct impacts of an SSC and indirect impact effects	PS4	The frequency of a meteor or satellite strike is judged to be so low as to make the risk impact from such events insignificant. Based on this review, the Meteorite or Satellite hazard can be considered to be negligible.

Hazard	Definition	Screening Criteria	Disposition for 10 CFR 50.69
	such as thermal effects (e.g., radiative heat transfer), overpressure effects, seismic effects, and the effects of ejecta resulting from a ground strike.		
Pipeline Accident	A release of hazardous material, a release of combustion products, an explosion, or the generation of missiles due to an accident involving the rupture of a pipeline carrying hazardous materials.	C3	IPEEE Section 5.5.2.2 (Reference [57]) details screening analysis of the natural gas pipeline to the Independence Power Station, which is located about 3 miles southwest of the plant. Explosions from this source would result in less than 1.0 psi overpressure at the site (the screening damage criterion). Other natural gas pipelines, such as to the Alcan Rolled Products Company, are further removed and of smaller diameter and thus will also pose no major risk (Reference [57]). See also Toxic Gas. Based on this review, the Pipeline Accident hazard can be considered to be negligible.

Hazard	Definition	Screening Criteria	Disposition for 10 CFR 50.69
Precipitation, Intense	Flooding that results from local intense precipitation. Secondary hazards resulting from local intense precipitation, include, but are not necessarily limited to, dam failure and river and stream overflow.	C1	The FHRR identified local intense precipitation (LIP) stillwater maximum water surface elevation (WSE) of 272.8 ft. This is greater than the existing CDB controlling flood elevation of 262 ft and is slightly above site grade, which is nominally 272 ft. In-leakage from exterior doors that are normally closed was evaluated in JAF-RPT-14-00035 (Reference [60]) to quantify the volume of water that could possibly enter any buildings from the 0.8 ft of ponding during the LIP event. The conclusions were that in-leakage will be minimal and interior drainage features would have more than enough capacity to mitigate the effects of any in-leakage from the normally closed exterior doors. Procedure AOP-13 (Reference [61]) directs operators to verify water intrusion is not occurring at building outer doors and to close doors as appropriate if sustained local intense precipitation is occurring. Otherwise, no human actions are required to mitigate the effects of a flooding event at the station. There are several doors whose failure to be in their normal position would result in an unscreened scenario. These doors are a mixture of personnel

Hazard	Definition	Screening Criteria	Disposition for 10 CFR 50.69
			doors, roll up doors, and vertical hatches that are required to be closed. Therefore, these doors and barriers should be categorized as HSS in accordance with NRC-approved guidance since removal of any of these doors would result in an unscreened scenario. With credit taken for the normally positioned personnel doors, external flooding mechanisms are screened using Criterion C1 and will not impact 10 CFR 50.69 categorization of other SSCs. See also External Flood.
Release of Chemicals from Onsite Storage	A release of hazardous material including, but not limited to liquids, combustion products, or radioactivity. Such releases may be concurrent with or induce an explosion or the generation of missiles. In this context, an onsite release of radioactivity is assumed to be associated with low-level radioactive waste.	C1 PS4	Per UFSAR Section 16.9.3.24 (Reference [63]), few chemicals which, when accidentally released, could pose safety hazard to the control room operators were evaluated using the VAPOR computer code. The results of this analysis have indicated that for the conditions existing at the plant site, none of these chemicals could produce conditions which would incapacitate control room operators (**C1**). See also Toxic Gas. IPEEE Section 5.5.2.2 (Reference [57]) details explosion hazards from chemicals onsite, including hydrogen, propane, and carbon-dioxide. All the explosion hazards for chemicals stored

Hazard	Definition	Screening Criteria	Disposition for 10 CFR 50.69
			onsite are screened based on negligible damage impact from the explosion **(C1)**, low frequency of occurrence **(PS4)**, or low CDF **(PS4)**. See also Toxic Gas. Based on this review, the Release of Chemicals in Onsite Storage hazard can be considered to be negligible.
River Diversion	The redirection of all or a portion of river flow by natural causes (e.g., a riverine embankment landslide) or intentionally (e.g., power production, irrigation).	C3	Per UFSAR Section 2.4.1 (Reference [63]), there are no naturally occurring, perennial streams on the site, but drainage ditches were constructed parallel to the western and eastern boundaries of the power block. Storm water runoff at the plant discharges to Lake Ontario via overland flow, intermittent streams, the drainage ditches, and/or the plant's storm drain system. Based on this review, the River Diversion hazard can be considered to be negligible.
Sandstorm	Persistent heavy winds transporting sand or dust that infiltrate SSCs at or in the plant's analyzed area causing functional failure.	C1	The plant is designed for such events. More common wind-borne dirt can occur but poses no significant risk to JAF given the robust structures and protective features of the plant. Based on this review, the Sand or Dust Storm hazard can be considered to be negligible.

Hazard	Definition	Screening Criteria	Disposition for 10 CFR 50.69
Seiche	Flooding from water displaced by an oscillation of the surface of a landlocked body of water, such as a lake, that can vary in period from minutes to several hours.	C1	The evaluation of the impact of the external flooding hazard at the site was updated as a result of the NRC's post Fukushima 50.54(f) Request for Information. The station's flood hazard reevaluation report (FHRR) was submitted to NRC for review (Reference [65]) and NRC issued an interim staff response letter for the station (Reference [66]) confirming the findings in the FHRR. The results indicate that all flood causing mechanisms, except LIP, PMF, and Combined Effects flooding are bounded by the current design basis (CDB) and do not pose a challenge to the plant. Based on this review (full details in Section 2.2), the Seiche hazard can be considered to be negligible.
Seismic Activity	Sudden ground motion or vibration of the Earth as produced by a rapid release of stored-up energy along an active fault. Secondary hazards resulting from seismic activity include, but are not necessarily limited to, avalanche (both rock and snow), dam failure, industrial accidents, landslide, seiche, tsunami, and vehicle accidents.	N/A	See Section 3.2.3 and Figure A4-1 in this Attachment.

Hazard	Definition	Screening Criteria	Disposition for 10 CFR 50.69
Snow	The accumulation of snow could result in transportation accidents or directly cause dynamic loading or freezing conditions as a result of snow cover.	C5	This hazard is slow to develop and can be identified via monitoring and managed via normal plant processes. Potential flooding impacts are accounted for in the External Flooding screening. Based on this review, the Snow hazard can be considered to be negligible.
Soil Shrink-Swell	Dynamic forces on structures' foundations due to the expansion (swelling) and contraction (shrinking) of soil resulting from changes in the soil moisture content.	C1 C5	The potential for this hazard is low at the site, the plant design considers this hazard (**C1**), and the hazard is slow to develop and can be mitigated (**C5**). Based on this review, the Soil Shrink-Swell Consolidation impact hazard can be considered to be negligible.
Storm Surge	Flooding that results from an abnormal rise in sea level due to atmospheric pressure changes and strong wind generally accompanied by an intense storm. Secondary hazards resulting from a storm surge include, but are not necessarily limited to, high tide, river and stream overflow, and waves.	C1	The evaluation of the impact of the external flooding hazard at the site was updated as a result of the NRC's post Fukushima 50.54(f) Request for Information. The station's flood hazard reevaluation report (FHRR) was submitted to NRC for review (Reference [65]) and NRC issued an interim staff response letter (Reference [66]) confirming the findings in the FHRR. The results indicate that all flood causing mechanisms, except LIP, PMF, and Combined Effects flooding are bounded by the current design basis (CDB) and do not pose a challenge to the plant.

Hazard	Definition	Screening Criteria	Disposition for 10 CFR 50.69
			Based on this review (full details in Section 2.2), the Storm Surge can be considered to be negligible.
Toxic Gas	A release of hazardous toxic or asphyxiant gases. Such releases may be concurrent with or induce an explosion or the generation of missiles. In this context, an onsite release of radioactivity is assumed to be associated with low-level radioactive waste.	C1	IPEEE Sections 5.5.1.2 and 5.5.2.1 (Reference [57]) discuss toxic chemical control room habitability studies. The toxic chemical release scenarios considered were from on-site chemical storage and offsite industrial facilities. The only shipments of potentially hazardous material transported within five miles of the site were those shipped to or from facilities within that distance, including the site. The calculations performed for the worst scenarios all concluded that the maximum predicted control room concentrations of toxic chemicals were all below the toxicity acceptance criteria. Therefore, there are no toxic gas hazards that require either gas detectors or automatic isolation of the control room. Based on this review, the Toxic Gas hazard can be considered to be negligible.
Transportation Accidents	Accidents involving transportation resulting in collision with SSCs, a release of hazardous materials or combustion products, an explosion, or a generation of	C3 C1 C4	IPEEE Section 5.5.2.1 (Reference [57]), shipments of hazardous material by boat, rail, and highway (except to or from local facilities) are not sources of concern because the shipping lane nearest to JAF is seven miles

Hazard	Definition	Screening Criteria	Disposition for 10 CFR 50.69
	missiles causing functional failure of SSCs. Hazards that could potentially result in transportation accidents include, for example, a vehicle, railcar or ship (boat) accident that involves a collision or derailment, potentially resulting in fire, explosions, toxic releases, missiles, or other hazardous conditions.		from the plant and serves the port of Oswego (this port mostly handles potash and urea; hazardous chemicals are not routinely handled at the port that lies nine miles from the plant); Conrail states that no hazardous chemicals are transported within five miles of the plant; and shipments of hazardous material by road, other than to or from local facilities, must use interstate highways (**C3**). Other releases of toxic gases during transportation to local facilities are included in the discussion of the Toxic Gas hazard screening (**C1**, **C4**). Based on this review, the Transportation Accident hazard can be considered to be negligible.
Tsunami	Flooding that results from a series of long-period sea waves that displaces massive amounts of water as a result of an impulsive disturbance, such as a major submarine slides or landslide. Secondary hazards resulting from a tsunami include, but are not necessarily limited to,	C3	The location of JAF along Lake Ontario precludes the possibility of a tsunami.

Hazard	Definition	Screening Criteria	Disposition for 10 CFR 50.69
	river and stream overflow.		
Turbine-Generated Missiles	Damage to safety-related SSCs from a missile generated internal or external to the plant PRA boundary from rotating turbines or other external sources (e.g., high-pressure gas cylinders). Damage may result from a falling missile or a missile ejected directly toward safety-related SSCs (i.e., low-trajectory missiles).	PS4	Per UFSAR Section 10.2.4 (Reference [63]), the probability of turbine missile generation has been evaluated and demonstrated to be in conformance with the NRC acceptance criterion of $1x10^{-4}$/yr for favorably oriented turbines such as the machine at JAF. For the monoblock rotors used at JAF, missile generation probability is based on the probability of a control system failure that results in an overspeed event in which turbine speed exceeds 120% of the normal (rated) operating speed along with demonstration of margin to ductile failure limits for rotating components. The probability of exceeding 120% of normal operating speed was calculated to be $4.35x10^{-5}$/yr. Per Regulatory Guide 1.115 Reference [67], this meets the acceptance criteria because the probability of turbine missile being generated and ejected from the casing is <1E-4/yr; since the probability of striking and damaging a safety-related (SR) component is 1E-3 (per Regulatory Guide 1.115), the frequency of damaging SR component is less than 1E-7/yr, so CDF is less than 1E-6/yr. Therefore, this hazard can be screened.

Hazard	Definition	Screening Criteria	Disposition for 10 CFR 50.69
			Based on this review, the Turbine-Generated Missiles hazard can be considered to be negligible.
Volcanic Activity	Opening of Earth's crust resulting in tephra (i.e., rock fragments and particles ejected by volcanic eruption), lava flows, lahars (i.e., mud flows down volcano slopes), volcanic gases, pyroclastic flows (i.e., fast-moving flow of hot gas and volcanic matter moving down and away from a volcano), and landslides. Indirect impacts include distant ash fallout (e.g., tens to potentially thousands of miles away). Secondary hazards resulting from volcanic activity, include, but are not necessarily limited to, seismic activity and fire.	C3	This hazard is not applicable to the site because of location (no active or dormant volcanoes located near plant site). Based on this review, the Volcanic Activity hazard can be considered to be negligible.
Waves	An area of moving water that is raised above the main surface of a body of water as a result of the wind blowing over an area of fluid surface.	C1 C4	The evaluation of the impact of the external flooding hazard at the site was updated as a result of the NRC's post Fukushima 50.54(f) Request for Information. The station's flood hazard reevaluation report (FHRR) was submitted to

Hazard	Definition	Screening Criteria	Disposition for 10 CFR 50.69
			NRC for review (Reference [65]), and NRC issued an interim staff response letter (Reference [66]) confirming the findings in the FHRR. The results indicate that all flood causing mechanisms, except LIP, PMF, and Combined Effects flooding are bounded by the current design basis (CDB) and do not pose a challenge to the plant (**C1**). Waves are considered in screening of the External Flooding hazard (**C4**). Based on this review, the impacts from waves hazard can be considered to be negligible and can be screened.

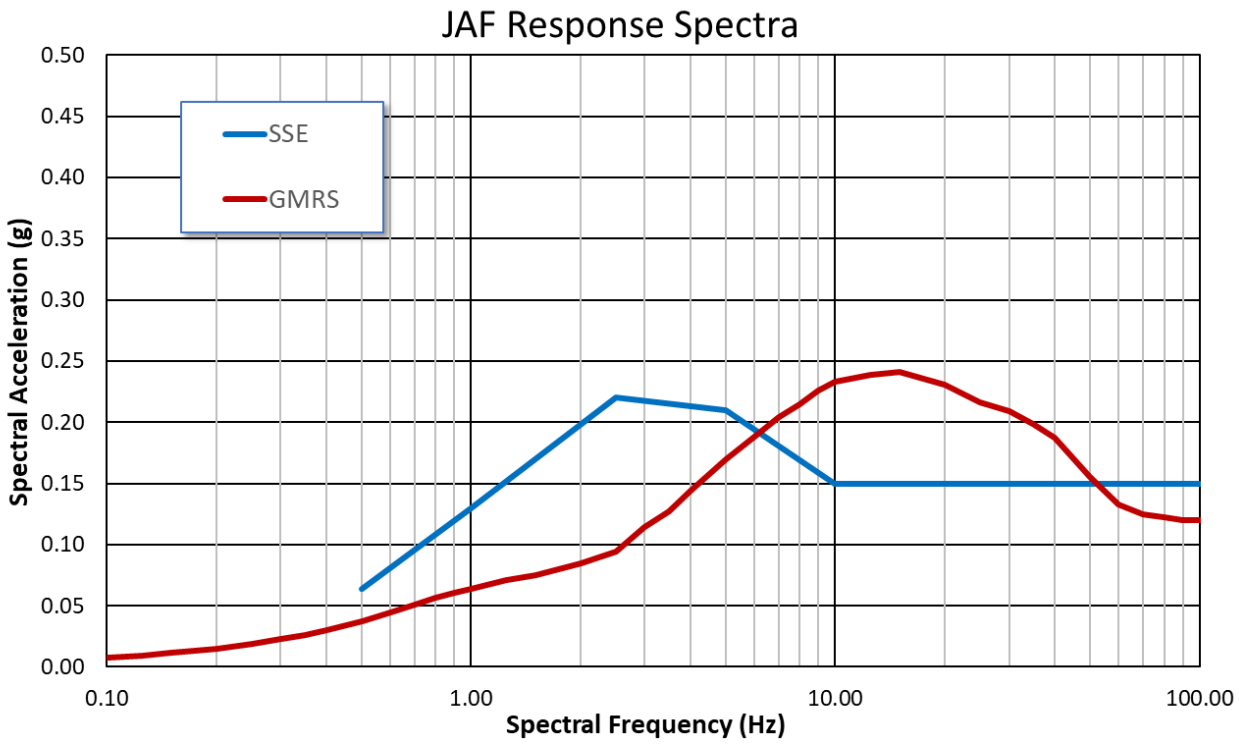

Figure A4-1: GMRS and SSE Response Spectra for JAF
(From Reference [26]).

Attachment 5: Progressive Screening Approach for Addressing External Hazards

Event Analysis	Criterion	Source	Comments
Initial Preliminary Screening	C1. Event damage potential is < events for which plant is designed.	NUREG/CR-2300 and ASME/ANS Standard RA-Sa-2009	
	C2. Event has lower mean frequency and no worse consequences than other events analyzed.	NUREG/CR-2300 and ASME/ANS Standard RA-Sa-2009	
	C3. Event cannot occur close enough to the plant to affect it.	NUREG/CR-2300 and ASME/ANS Standard RA-Sa-2009	
	C4. Event is included in the definition of another event.	NUREG/CR-2300 and ASME/ANS Standard RA-Sa-2009	
	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	

Event Analysis	Criterion	Source	Comments
	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	

Attachment 6: Disposition of Key Assumptions/Sources of Uncertainty

Assessment of Internal Events PRA Epistemic Uncertainty Impacts

In order to identify key sources of uncertainty, the Internal Events baseline PRA model uncertainty report was developed, based on the guidance in NUREG-1855 (Reference [68]) and EPRI 1016737 (Reference [42]). As described in NUREG-1855, sources of uncertainty include "parametric" uncertainties, "modeling" uncertainties, and "completeness" (or scope and level of detail) uncertainties.

Parametric uncertainty was addressed as part of the JAF baseline PRA model quantification. The parametric uncertainty evaluation for the Internal Events PRA model is documented in Appendix B of the Summary Notebook (Reference [69]).

Modeling uncertainties are considered in both the base PRA and in specific risk-informed applications. Assumptions are made during the PRA development as a way to address a particular modeling uncertainty because there is not a single definitive approach. Plant-specific assumptions made for each of the JAF Internal Events PRA technical elements are noted in the Summary Notebook (Reference [69]). The Internal Events PRA model uncertainties evaluation considers the modeling uncertainties for the base PRA by identifying assumptions, determining if those assumptions are related to a source of modeling uncertainty and characterizing that uncertainty, as necessary. The Electric Power Research Institute (EPRI) compiled a listing of generic sources of modeling uncertainty to be considered for each PRA technical element (Reference [42]), and the evaluation performed for JAF considered each of the generic sources of modeling uncertainty as well as the plant-specific sources.

Completeness uncertainty addresses scope and level of detail. Uncertainties associated with scope and level of detail are documented in the PRA but are only considered for their impact on a specific application. No specific issues of PRA completeness have been identified relative to the 10 CFR 50.69 application, based on the results of the Internal Events PRA and Fire PRA peer reviews.

The impact of potential sources of uncertainty for the Internal Events model on the PRA or applications is discussed in the table below.

> *Note*: As part of the required 10 CFR 50.69 PRA categorization sensitivity cases directed by NEI 00-04, internal events / internal flood and fire PRA models' human error and common cause basic events are increased to their 95th percentile and also decreased to their 5th percentile values. These results are capable of driving a component and respective functions HSS and therefore the uncertainty of the PRA modeled HEPs and CCFs are accounted for in the 10 CFR 50.69 application.

Disposition of IE/IF PRA Key Assumptions/Sources of Uncertainty

IE / IF PRA Sources of Assumption/ Uncertainty	IE / IF PRA 10 CFR 50.69 Impact	IE / IF PRA Model Sensitivity and Disposition (10 CFR 50.69)
Conditional probability that drywell shell melt-through occurs despite the availability of water injection to the debris.	The ability to prevent melt-through of the steel containment shell is dependent upon the availability of water to cool the core debris. For cases in which there is limited or no water available to the debris, the containment is modeled as likely to fail.	A sensitivity study was performed which demonstrated that the FPIE and FPRA LERF (using the average maintenance PRA models) are not sensitive to the failure probability associated with drywell shell melt-through.
Relay Room Internal Flooding scenarios conservatively assume complete failure of all equipment if water level reaches 9". In reality, equipment located higher in panels may survive and support one or more success paths.	The operator action for "Operator Fails to Perform Shutdown Outside Control Room for Relay Room Flood" was both increased and decreased by one decade in separate quantifications. This operator action is not highly uncertain, but was judged reasonable to use for the sensitivity because it is included in various model logic elements for Relay Room Internal Flooding.	The results demonstrate that FPIE CDF and LERF are highly sensitive to an increased probability that operators fail to perform plant shutdown following a Relay Room flood; however, this increased probability would not be realistic and would likely mask key risk insights. In addition, as part of the required 10 CFR 50.69 PRA categorization sensitivity cases directed by NEI 00 04, internal events / internal flood modeled human error basic events are increased to their 95th percentile and also decreased to their 5th percentile values. These results are capable of driving a component and respective functions HSS and therefore the uncertainty of the PRA modeled HEPs are accounted for in the 10 CFR 50.69 application.

IE / IF PRA Sources of Assumption/ Uncertainty	IE / IF PRA 10 CFR 50.69 Impact	IE / IF PRA Model Sensitivity and Disposition (10 CFR 50.69)
Conditional LOOP probabilities are based on a relatively few events with some “partial failure” experience, which could be biased toward higher failure probabilities.	The probabilities that a conditional LOOP occurs after a transient and after a LOCA are increased and reduced by a factor of 2 in separate quantifications.	The results demonstrate that the conditional LOOP probabilities do not significantly impact the overall average maintenance FPIE and FPRA results.
The likelihood of containment failure below drywell (DW) head elevation for CRD injection capability is potentially non-conservative given the unknown phenomenological events associated with containment venting that would affect equipment survivability.	The probability that containment failure occurs below the DW head elevation is both increased and reduced in separate quantifications.	The results demonstrate that the probability of containment failure occurring below DW head elevation does not significantly impact the overall average maintenance PRA results.
Spurious opening of circuit breakers is modeled as a credible means for de-energization of credited equipment; however, this failure mode is among the potentially more recoverable because such breakers often reclose successfully when manipulated by operators.	Slight conservatism considering potential impact of recovery of spurious circuit breaker operation. The failure rates for spurious operation of circuit breakers were reduced by one decade to simulate a chance for successful recovery.	The results demonstrate that credit for equipment repair and recovery does not impact the overall average maintenance PRA results.
Little credit is taken for the Reactor Building as an effective means of fission product retention due to the reliance on the Standby Gas Treatment System and potential for hydrogen	The PRA currently takes little credit (e.g., failure rate = 0.99) for secondary containment fission product retention capability to prevent a large and early release. The probabilities that the Reactor Building is	The results demonstrate that FPIE and FPRA LERF are highly sensitive to changes in Reactor Building effectiveness. However, this sensitivity analysis assumes the Reactor Building fails with a low probability, which is not

IE / IF PRA Sources of Assumption/ Uncertainty	IE / IF PRA 10 CFR 50.69 Impact	IE / IF PRA Model Sensitivity and Disposition (10 CFR 50.69)
deflagration during severe accidents.	ineffective given DW or torus airspace failures, drywell shell melt-through, and torus below water line failure were each adjusted to 0.1.	considered realistic in the PRA due to significant challenges in achieving Reactor Building fission product retention during a severe accident that would potentially lead to a large and early release. There remains little justifiable basis to credit the Reactor Building for significant attenuation of releases.
The likelihood that SRVs fail open in a severe accident is credited somewhat conservatively in the PRA due to limitations on experience with SRV response in severe accidents and phenomenological analysis.	The current model provides the most realistic assessment defensible, but does rely on judgment because experience with SRV response in severe accidents is limited and phenomenological analysis is likewise limited. A value of 0.248 is used in the PRA and, while realistic, is likely some degree of conservative. The probabilities for failure to depressurize the RPV in Class IA and IBE sequences were adjusted to 0.1.	The results demonstrate that the depressurization failure probabilities have a slight impact on FPIE and FPRA LERF. The lower bound probabilities for failure to depressurize are difficult to assess as realistic due to limited experience with SRV response in severe accidents. As part of the required 10 CFR 50.69 PRA categorization sensitivity cases directed by NEI 00 04, internal events / internal flood and fire PRA models' human error and common cause basic events are increased to their 95th percentile and also decreased to their 5th percentile values. These results are capable of driving a component and respective functions HSS and therefore the uncertainty of the PRA modeled HEPs and CCFs are accounted for in the 10 CFR 50.69 application

IE / IF PRA Sources of Assumption/ Uncertainty	IE / IF PRA 10 CFR 50.69 Impact	IE / IF PRA Model Sensitivity and Disposition (10 CFR 50.69)
FLEX equipment failure probabilities are identified as a candidate source of model uncertainty since there are no industry-approved data sources for FLEX equipment reliability.	FLEX equipment is not yet explicitly modeled in the JAF PRA; however, there are no industry sources to provide FLEX equipment reliability data. The PRA currently models the FLEX diesel generator with an HEP-based basic event based on human action being taken as the dominant contributor for FLEX generator failure probability.	A sensitivity study was performed which demonstrated that FLEX equipment failure does not significantly impact the overall average maintenance PRA results.

Assessment of Supplementary FPRA Epistemic Uncertainty Impacts

The purpose of the following discussion is to address the epistemic uncertainty in the JAF FPRA. The FPRA model includes various sources of uncertainty that exist because there is both inherent randomness in elements that comprise the FPRA, and because the state of knowledge in these elements continues to evolve. The development of the FPRA was guided by NUREG/CR-6850 (Reference [70]). The FPRA model used consensus models described in NUREG/CR-6850. In order to identify key sources of uncertainty for the 10 CFR 50.69 Program Application, an evaluation of Fire PRA model uncertainty was performed, based on the guidance in NUREG-1855 (Reference [68]) and Electric Power Research Institute (EPRI) report 1026511 (Reference [71]).

As stated in Section 1.3 of NUREG-1855:

> "Although the guidance in the this [sic] report does not currently address all sources of uncertainty, the guidance provided on the uncertainty identification and characterization process and on the process of factoring the results into the decision making is generic and independent of the specific source of uncertainty. Consequently, the guidance is applicable for sources of uncertainty in PRAs that address at-power and low power and shutdown operating conditions, and both internal and external hazards."

NUREG-1855 also describes an approach for addressing sources of model uncertainty and related assumptions. It states:

> "A source of model uncertainty exists when (1) a credible assumption (decision or judgment) is made regarding the choice of the data, approach, or model used to address an issue because there is no consensus and (2) the choice of alternative data, approaches or models is known to have an impact on the PRA model and results. An impact on the PRA model could include the introduction of a new basic event, changes to basic event probabilities, change in success criteria, or introduction of a new initiating event. A credible assumption is one submitted by relevant experts and which has a sound technical basis. Relevant experts include those individuals with explicit knowledge and experience for the given issue. An example of an assumption related to a source of model uncertainty is battery depletion time. In calculating the depletion time, the analyst may not have any data on the time required to shed loads and thus may assume (based on analyses) that the operator is able to shed certain electrical loads in a specified time."

Section 2.1.3 of NUREG-1855 defines consensus model as:

> "A consensus model is a model that has a publicly available published basis and has been peer reviewed and widely adopted by an appropriate stakeholder group. In addition, widely accepted PRA practices may be regarded as consensus models. Examples of the latter include the use of the constant probability of failure on demand model for standby components and the Poisson model for initiating events. For risk-informed regulatory decisions, the consensus model approach is one that the NRC has used or accepted for the specific risk-informed application for which it is proposed."

Modeling uncertainties are considered in both the base Fire PRA and in specific risk-informed applications. Assumptions are made during the Fire PRA development as a way to address a particular modeling uncertainty because there is not a single definitive approach. Plant-specific assumptions made for each of the JAF Fire PRA technical elements are noted in the Uncertainty and Sensitivity Analysis Notebook (Reference [72]).

Completeness uncertainty addresses scope and level of detail. Uncertainties associated with scope and level of detail are documented in the Fire PRA, but are only considered for their impact on a specific application (Reference [73]). No specific issues of PRA completeness have been identified relative to the 10 CFR 50.69 application, based on the results of the Internal Events PRA and Fire PRA peer reviews.

EPRI compiled a listing of generic sources of modeling uncertainty to be considered for each Fire PRA technical element in EPRI report 1026511 (Reference [71]). Based on following the methodology in EPRI 1026511 for a review of sources of uncertainty, the impact of potential sources of uncertainty on the PRA or applications was performed.

The table below describes the fire PRA sources of model uncertainty and their impact for the 10 CFR 50.69 Application relative to each task of NUREG/CR-6850 (Reference [70]).

Disposition of FPRA Key Assumptions/Sources of Uncertainty

Fire PRA Description	Fire PRA Sources of Uncertainty	Fire PRA Disposition
Plant Boundary Definition and Partitioning	No plant specific items relative to plant partitioning.	Consistent with NUREG/CR-6850. The results of the plant partitioning task are evaluated via fire scenarios and the multi-compartment analysis. This does not represent a key source of uncertainty for the JAF 10 CFR 50.69 Application.
Equipment Component Selection	No plant specific items relative to Equipment Selection. The Fire PRA assumes at a minimum a plant trip. This is consistent with accepted industry practice. The Fire PRA does not credit certain FPIE equipment or systems. Systems are included based on an iterative process to include equipment that may be significant to the fire risk.	Uncertainty exists in the assumed plant trip and the scope of equipment credited in the Fire PRA. Sensitivity evaluation is performed for equipment or systems not credited in the Fire PRA. There is no current accepted industry guidance to evaluate the plant trip probability. This does not represent a key source of uncertainty for the JAF 10 CFR 50.69 Application.
Cable Selection	No plant specific items relative to Cable Selection. The cable selection task for JAF includes accepted industry practice for circuit analysis. Fire PRA cables identified without known routing locations are assigned routing locations based on the available data. Therefore, the uncertainty related to these routings is small. The assumptions related to the modeling of lack of coordination may be conservative. Lastly, certain equipment and systems were not credited in the Fire PRA.	Uncertainty exists for cables selected specifically for the Fire PRA and have missing cable routing location data. Fire PRA assigns routing locations based on known routing locations. Therefore, this is not identified as a key source of uncertainty. Also, the Fire PRA includes a conservative treatment of the lack of coordination. This treatment is refined as needed per fire scenario. This does not represent a key source of uncertainty for the JAF 10 CFR 50.69 Application.

Fire PRA Description	Fire PRA Sources of Uncertainty	Fire PRA Disposition
Qualitative Screening	No plant specific items relative to Qualitative Screening. The plant partitioning, equipment selection, and cable selection task are performed to ensure with confidence the Fire PRA does not screen plant locations that may contribute to the fire risk.	This does not represent a key source of uncertainty for the JAF 10 CFR 50.69 Application.
Plant Response Modell	No plant specific items relative to Plant Response Model. The Internal Events and Fire PRA use the same modeling and fire induced failures are added. The Fire PRA uses the same data used in the FPIE model. Fire induced failures use accepted industry guidance. The Fire PRA models the MCRAB sequence consistent with the equipment available for remote shutdown and industry guidance for remote shutdown operator actions.	The Internal Events PRA identifies plant specific uncertainties which are also applicable to the Fire PRA and should be included for applications, as applicable. These uncertainties include HRA and CCF treatment. Other uncertainties may be applicable; for example, the amount of credit for injection sources. Additionally, the uncertainties associated with the basic event data are identified in the JAF parametric uncertainty analysis. This does not represent a key source of uncertainty for the JAF 10 CFR 50.69 Application.
Fire Ignition Frequency	No plant specific items relative to Fire Ignition Frequency. The key uncertainties related to fire ignition frequency remain related to the generic industry data accepted for use.	The generic fire frequencies have been identified as the main source of uncertainty in this task. NUREG-2169 has collected additional data which has improved some of these issues, as supplemented by subsequent guidance. However, the same uncertainty issues exist in NUREG-2169 and thus the characterization and treatment of uncertainty in the Fire PRA is the same as NUREG/CR-6850. In summary, the generic frequencies

Fire PRA Description	Fire PRA Sources of Uncertainty	Fire PRA Disposition
		are estimated and uncertainty bounds are provided. These uncertainty bounds account for the inherent randomness of the occurrence of fire events, and the variability among the plants. Fire ignition frequency distributions are propagated in the JAF parametric uncertainty analysis. This does not represent a key source of uncertainty for the JAF 10 CFR 50.69 Application.
Quantitative Screening	No plant specific items relative to Quantitative Screening.	Quantified Physical Analysis Units scenarios were maintained. This does not represent a key source of uncertainty for the JAF 10 CFR 50.69 Application.
Fire Scoping Modeling	No plant specific items relative to Fire Scoping Model.	This does not represent a key source of uncertainty for the JAF 10 CFR 50.69 Application.
Detailed Circuit Failure Analysis	See Fire PRA Cable Selection Task of this table.	See Fire PRA Cable Selection Task of this table.
Circuit Failure Model Likelihood Analysis	No plant specific items relative to Circuit Failure Mode Likelihood Analysis.	The uncertainties are related to the specific configuration of the component cables being analyzed and the generic probabilities applied. Circuit failure probabilities are propagated in the parametric uncertainty evaluation. This does not represent a key source of uncertainty for the JAF 10 CFR 50.69 Application.
Fire Scenario Selection and Fire Modeling	No plant specific items relative to Fire Scenario Selection. The fire scenario selection and fire models use accepted industry guidance. Fire scenarios are	The fire scenario selection and fire modeling is iterative such that the fire risk contribution from fire ignition sources and PAUs is reasonable and realistically characterized based on

Fire PRA Description	Fire PRA Sources of Uncertainty	Fire PRA Disposition
	generally developed in iterative stages. A conservative definition is initially applied and the range of accepted fire modeling methods and tools are iteratively applied such that the resulting fire risk contributors are reasonable based on the accepted methods.	the accepted guidance, methods, and JAF specific characteristics. Plant specific inputs are used when available and otherwise supplemented by industry accepted data. The fire scenario SF and NSP uncertainties are estimated in the parametric uncertainty evaluation. This does not represent a key source of uncertainty for the JAF 10 CFR 50.69 Application.
Human Reliability Analysis	Plant specific key assumptions include conservative adjustments made to account for fire events. Additionally, a JHEP floor value of 1E-6 is applied. Assumptions related to fire event impacts on operator actions are conservatively applied and refined as needed with operator training personnel input. A JHEP floor value is used and is considered to be good practice.	The Fire PRA includes conservative adjustments to the HFEs to account for adverse impacts of fire events. The Fire PRA does not include credit for all operator actions, including fire response actions. The Fire PRA does not include credit for all of the instrument cues that may be available. A minimum joint HEP was applied for the HRA dependency analysis. Applying a minimum joint HEP may skew the results by artificially increasing the risk due to human actions. The HEPs are propagated in the parametric uncertainty evaluation based on the uncertainty parameters from the HRAC.As part of the required 50.69 PRA categorization sensitivity cases directed by NEI 00-04, Internal Events / Internal Flood and Fire PRA models' human error basic events are increased to their 95th percentile and also decreased to their 5th percentile values. These results are capable of driving a component and respective functions HSS and therefore, the

Fire PRA Description	Fire PRA Sources of Uncertainty	Fire PRA Disposition
		uncertainty of the PRA modeled HEPs are accounted for in the 50.69 Application. This does not represent a key source of uncertainty for the JAF 10 CFR 50.69 Application.
Seismic-Fire Interactions Assessment	No plant specific items relative to Seismic Fire.	This does not represent a key source of uncertainty for the JAF 10 CFR 50.69 Application.
Fire Risk Quantification	No plant specific items relative to Fire Quantification. The Fire PRA is quantified using accepted industry codes and practices. The fire quantification task does identify truncation limit as a source of uncertainty; however, this was investigated in truncation studies to ensure an appropriate truncation limit was used.	Technical reviews address the uncertainties identified in each task to confirm adequacy for estimating the fire risk. Truncation study used to ensure risk contributors were not inadvertently omitted from the results. This does not represent a key source of uncertainty for the JAF 10 CFR 50.69 Application.