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10 CFR 50.90
10 CFR 50.69

May 20, 2021

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

R. E. Ginna Nuclear Power Plant
Renewed Facility Operating License No. DPR-18
NRC Docket No. 50-244

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 R. E. Ginna Nuclear Power Plant (Ginna).

The proposed amendment would modify the Ginna 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 Ginna 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 May 20, 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,'" (ML21140A324). 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 May 20, 2022, with the amendment being implemented within 60 days following NRC approval.

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

Attachment 7 contains a summary of commitments.

Should you have any questions concerning this submittal, please contact Jessie Hodge at (610) 765-5532.

I declare under penalty of perjury that the foregoing is true and correct. Executed on this 20th day of May 2021.

Respectfully,

David T. Gudger

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

Enclosure: 1. Evaluation of the Proposed Change

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

Enclosure
Evaluation of the Proposed Change

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1 SUMMARY DESCRIPTION

The proposed amendment modifies the licensing basis to allow for the implementation of the provisions of Title 10 of the Code of Federal Regulations (10 CFR), Part 50.69, "Risk-Informed Categorization and Treatment of Structures, Systems and Components for Nuclear Power Reactors." The provisions of 10 CFR 50.69 allow adjustment of the scope of equipment subject to special treatment controls (e.g., quality assurance, testing, inspection, condition monitoring, assessment, and evaluation). For equipment determined to be of low safety significance (LSS), alternative treatment requirements can be implemented in accordance with this regulation. For equipment determined to be of high safety significance (HSS), 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 Ginna 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 [May 20, 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 date, "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,'" (ML21140A324).

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

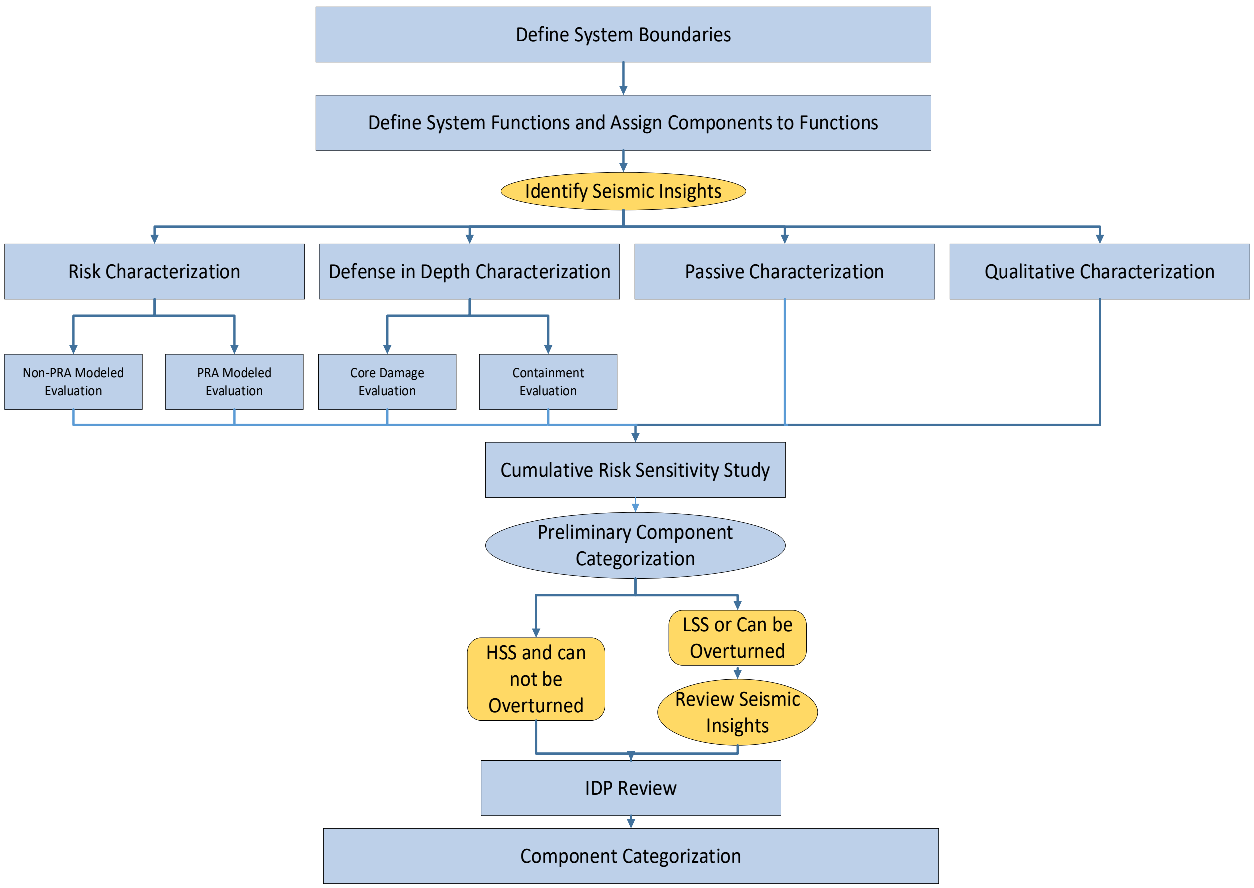
Exelon will implement the risk categorization process in accordance with NEI 00-04, Revision 0, as endorsed by Regulatory Guide (RG) 1.201, "Guidelines for Categorizing Structures, Systems, and Components in Nuclear Power Plants According to their Safety Significance" (Reference [2]). NEI 00-04 Section 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 1 sites, which includes Ginna, to assess seismic hazard risk for 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 complete, 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, Seismic Safe Shutdown Equipment List, 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., HSS or LSS that is presented to the Integrated Decision-Making Panel (IDP). Note: the term "preliminary HSS or LSS" is synonymous with the NEI 00-04 term "candidate HSS or LSS." A component or function is preliminarily categorized as HSS if any element of the process results in a preliminary HSS determination in accordance with Table 3-1 below. The safety significance determination of each element, identified above, is independent of each other and therefore the sequence of the elements does not impact the resulting preliminary categorization of each component or function. Consistent with NEI 00-04, the categorization of a component or function will only be "preliminary" until it has been confirmed by the IDP. Once the IDP confirms that the categorization process was followed appropriately, the final RISC category can be assigned.

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

Table 3-1: Categorization Evaluation Summary

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

Element	Categorization Step - NEI 00-04 Section	Evaluation Level	IDP Change HSS to LSS	Drives Associated Functions
	Containment – Section 6.2	Component	Not Allowed	Yes
Qualitative Criteria	Considerations – Section 9.2	Function	Allowable[1]	N/A
Passive	Passive – Section 4	Segment/Component	Not Allowed	No

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

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

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

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

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

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

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

- The IDP will be trained in the specific technical aspects and requirements related to the categorization process. Training will address at a minimum the purpose of the categorization; present treatment requirements for SSCs including requirements for 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 safety significant and LSS.

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

- An unreliability factor of 3 will be used for the 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 SE (Reference [4]) which states "…if any SSC is identified as HSS from either the integrated PRA component safety significance assessment (Section 5 of NEI 00-04) or the defense-in-depth assessment (Section 6), the associated system function(s) would be identified as HSS."

- Once a system function is identified as HSS, then all the components that support that function are preliminary HSS. The IDP must intervene to assign any of these HSS Function components to LSS.

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

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

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

- Internal Event Risks: Internal events including internal flooding PRA, as submitted to the NRC for TSTF 505 dated May 20, 2021, (ML21140A324) (Refer to Attachment 2).

- Fire Risks: Fire PRA model, as submitted to the NRC for TSTF 505 dated May 20, 2021, (ML21140A324) (Refer to Attachment 2).

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

- Other External Risks (e.g., tornados, external floods): Using the IPEEE screening process as approved by NRC SE dated December 21, 2000, (TAC No. M83624). 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 [5]), 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 [6] (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 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 Ginna 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 May 20, 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,'" (ML21140A324).

3.2.1 Internal Events and Internal Flooding

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

3.2.2 Fire Hazards

The Ginna 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 Ginna. 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 Ginna seismic hazard assessment, Ginna 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 Electric Power Research Institute (EPRI) 3002017583, "Alternative Approaches for Addressing Seismic Risk in 10 CFR 50.69 Risk-Informed Categorization,"(Reference [3]) and includes additional qualitative considerations that are discussed in this section[1]. Ginna meets the EPRI 3002017583 Tier 1 criteria for a "Low Seismic Hazard/High Seismic Margin" site. The Tier 1 criteria are as follows:

> *"Tier 1: Plants where the GMRS [Ground Motion Response Spectrum] peak acceleration is at or below approximately 0.2g or where the GMRS is below or approximately equal to the SSE [Safe Shutdown Earthquake] between 1.0 Hz and 10 Hz. Examples are shown in Figures 2-1 and 2-2. At these sites, the GMRS is either very low or within the range of the SSE such that unique seismic categorization insights are not expected."*
>
> > Note: EPRI 3002017583 applies to the Tier 1 sites in its entirety except for sections 2.3 (Tier 2 sites), 2.4 (Tier 3 sites), Appendix A (seismic correlation), and Appendix B (criteria for capacity-based screening).

The Tier 1 criterion (i.e., basis) in EPRI 3002017583 is a comparison of the ground motion response spectrum (GMRS, derived from the seismic hazard) to the safe shutdown earthquake (SSE, i.e., seismic design basis capability). U.S. nuclear power plants that utilize the 10 CFR 50.69 Seismic Alternative (EPRI 3002017583) will continue to compare GMRS to SSE.

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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 [65]) 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 [66]).

(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 [67]), 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.

The trial studies in EPRI 3002017583 show that seismic categorization insights are overlaid by other risk insights even at plants where the GMRS is far beyond the seismic design basis. Therefore, the basis for the Tier 1 classification and resulting criteria is not that the design basis insights are adequate. Instead, it is that consideration of the full range of the seismic hazard produces limited unique insights to the categorization process. That is the basis for the following statements in Table 4-1 of the EPRI report.

> *"At Tier 1 sites, the likelihood of identifying a unique seismic condition that would cause an SSC to be designated HSS is very low.*
>
> *Therefore, with little to no anticipated unique seismic insights, the 50.69 categorization process using the FPIE PRA and other risk evaluations along with the required Defense-in-Depth and IDP qualitative considerations are expected to adequately identify the safety-significant functions and SSCs required for those functions and no additional seismic reviews are necessary for 10 CFR 50.69 categorization."*

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

For example, the 10 CFR 50.69 categorization process as defined in NEI 00-04 includes an Integral Assessment that weighs the hazard specific relative importance of a component (e.g., internal events, internal fire, seismic) by the fraction of the total Core Damage Frequency (CDF) contributed by that hazard. The risk from an external hazard can be reduced from the default condition of HSS if the results of the integral assessment meets the importance measure criteria for LSS. For Tier 1 sites, the seismic risk (CDF/LERF) will be low such that seismic hazard risk is unlikely to influence an HSS decision. In applying the EPRI 3002017583 process for Tier 1 sites to the Ginna 10 CFR 50.69 categorization process, the IDP will be provided with the rationale for applying the EPRI 3002017583 guidance and informed of plant SSC-specific seismic insights for their consideration in the HSS/LSS deliberations.

EPRI 3002017583 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 EPRI report. The coupling of these concepts with the categorization process in NEI 00-04 are the key elements of the approach defined in EPRI 3002017583 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 [7]) provide inherent seismic capacities for most SSCs that are not directly related to the site-specific seismic demand. This inherent seismic capacity is based on the non-seismic design loads (pressure, thermal, dead weight, etc.) and the required functions for the SSC. For example, a pump has a relatively high inherent seismic capacity based on its design and that same seismic capacity applies at a site with a very low demand and at a site with a very high demand. At sites with lower seismic demands such as Ginna, there is no need to perform more detailed evaluations to demonstrate the inherent seismic capacities documented in industry sources such as Reference [7]. Low seismic demand sites have lower likelihood of seismically-induced failures and lesser challenges to plant systems. This, therefore, provides the technical basis for allowing use of a graded approach for addressing seismic hazard at Ginna.

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

The following provides the basis for establishing Tier 1 criteria in EPRI 3002017583.

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

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

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

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

Test cases described in Section 3 of Reference [3] showed that it would be unusual even for moderate hazard plants to exhibit any unique seismic insights, including due to correlated failures. The plant specific Reference [3] test case information Exelon is using from other licensees and being incorporated by Reference into this application is described in Case Study A (References [8], [9], and [10]), Case Study C (References [11], [12]), and Case Study D (References [13], [14], [15], [16], and [17]). Hence, while it is prudent to perform additional evaluations to identify conditions where correlated failures may occur for Tier 2 sites, for Tier 1 sites such as Ginna, correlation studies would not lead to new seismic insights or affect the baseline seismic CDF in any significant way.

The Tier 1 to Tier 2 threshold as defined in EPRI 3002017583 provides a clear and traceable boundary that can be consistently applied plant site to plant site. Additionally, because the boundary is well defined, if new information is obtained on the site hazard, a site's location within a particular Tier can be readily confirmed. In the unlikely event that the Ginna seismic hazard changes to medium risk (i.e., Tier 2) at some future time, Ginna will follow its categorization review and adjustment process procedures to review the changes to the plant and update, as appropriate, the SSC categorization in accordance with 10 CFR 50.69(e).

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

In response to the NRC 50.54(f) letter associated with post-Fukushima recommendations (Reference [18]), Ginna submitted a seismic hazard screening report (Reference [19]) to the NRC. The GMRS for Ginna is below the SSE between 1 Hz and 10 Hz and therefore meets the Tier 1 criterion in Reference [3].

The Ginna SSE and GMRS curves from the seismic hazard and screening response in Reference [19] are shown in Figure 1. The NRC's staff assessment of the Ginna seismic hazard and screening response is documented in Reference [20]. In section 3.4 of Reference [20] the NRC concluded that the methodology used by Ginna in determining the GMRS was acceptable and that the GMRS determined by Ginna adequately characterizes the reevaluated hazard for the Ginna 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 Ginna, 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 [19] [20]).
2. NTTF Recommendation 2.3 seismic walkdowns (References [21], [22])
3. NTTF Recommendation 4.2 seismic mitigation strategy assessment (S-MSA) (References [23], [24])

The following additional post-Fukushima seismic reviews were performed for Ginna.

4. NTTF Recommendation 2.1 seismic high frequency evaluation (References [25], [26])

The small percentage contribution of seismic to total plant risk makes it unlikely that an integral importance assessment for a component, as defined in NEI 00-04, would result in an overall HSS determination. Further, the low hazard relative to plant seismic capability makes it unlikely that any unique seismic condition would exist that would cause an SSC to be designated HSS for a Tier 1 site such as Ginna.

As an enhancement to the EPRI study results as they pertain to Ginna, the proposed Ginna categorization approach for seismic hazards will include qualitative consideration of the mitigation capabilities of SSCs during seismically-induced events and seismic failure modes, based on insights obtained from prior seismic evaluations performed for Ginna. For example, as part of the categorization team's preparation of the System Categorization Document (SCD) that is presented to the IDP, a section will be included in the SCD that summarizes the identified plant seismic insights pertinent to the system being categorized, and will also state the basis for applicability of the EPRI 3002017583 study and the bases for Ginna being a Tier 1 plant. The discussion of the Tier 1 bases will include such factors as:

- The low seismic hazard for the plant, which is subject to periodic reconsideration as new information becomes available through industry evaluations; and
- The definition of Tier 1 in the EPRI study.

At several steps of the categorization process (e.g., as noted in Figure 3-1 and Table 3-1) 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 Ginna) 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 Ginna 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 Ginna plant-specific seismic reviews and other resources such as those identified above. The objective is to identify plant-specific seismic insights derived from the above sources, relevant to the components in the system being categorized, that might include potentially important impacts such as:

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

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

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

Based on the above, the Summary/Conclusion/Recommendation from Section 2.2.3 of Reference [3] applies to Ginna, i.e., Ginna is a Tier 1 plant for which the GMRS is very low such that unique seismic categorization insights are expected to be minimal. As discussed in Reference [3], the likelihood of identifying a unique seismic insight that would cause an SSC to be designated HSS is very low. References [27], [28], and [29] are incorporated into this LAR as they provide additional supporting bases for Tier 1 plants. Therefore, with little to no anticipated unique seismic insights, the 10 CFR 50.69 categorization process using the FPIE PRA and other risk evaluations along with the defense-in-depth and qualitative assessment by the IDP adequately identify the safety-significant functions and SSCs.

3.2.4 Other External Hazards

All external hazards, except for seismic, were screened for applicability to Ginna per a plant-specific evaluation in accordance with GL 88-20 (Reference [30]) 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 Ginna 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.

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 Ginna. 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 [31]). The process in these References was mostly developed to evaluate the uncertainties associated with the internal events PRA model; however, the approach can be applied to other types of hazard groups.

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

Key Ginna 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 Ginna 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 [32]), consistent with NRC RIS 2007-06. The internal events model was assessed against RG 1.200, Revision 1, as discussed below. Additional information on the review of the internal events model against RG 1.200, Revision 2 was included as a response to RAI 1 in Section 3.1.4.1 of the Surveillance Frequency Control Program Safety Evaluation included in the ML16125A485.

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 [33]) as accepted by NRC in the letter dated May 3, 2017 (Reference [34]). The results of this review have been documented and are available for NRC audit.

Full Power Internal Events and Internal Flooding PRA Model

The Ginna FPIE PRA model was peer reviewed in June 2009 using the NEI 05-04 process, the PRA Standard (ASME RA-Sc-2007) and Regulatory Guide 1.200, Revision 1. This Peer Review (Reference [35] was a full-scope review of the technical elements of the Internal Events and internal flooding, at-power PRA.

In June 2017, an F&O Closure Review was conducted for the Ginna FPIE PRA model to evaluate elements of the PRA relative to the requirements of ASME/ANS RA-Sa-2009 and RG 1.200 Rev. 2 (Reference [32].

In January 2020, a second F&O Closure Review was conducted for the Ginna PRA Model (Reference [36]). The Internal Events scope of the review was the open and partially resolved finding-level F&Os from the 2017 F&O Closure Review (Reference [37]. The focused-scope peer review determined there is one finding-level F&O that remains open resulting in a Capability Category I SR. This finding level F&O is discussed in Attachment 3.

Fire PRA Model

The Ginna Fire PRA (FPRA) peer review (Reference [38]) was performed in June 2012 using the NEI 07-12 Fire PRA peer review process (Reference [39]), the ASME PRA Standard, ASME/ANS RA-Sa-2009 (Reference [40]), and Regulatory Guide 1.200, Rev. 2 (Reference [32]). The purpose of this review was to establish the technical acceptability of the FPRA for the spectrum of potential risk-informed plant licensing applications for which the FPRA may be used. The FPRA peer review was a full-scope review of all of the technical elements of the Ginna 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 findings from the Fire PRA peer review have been addressed in the Fire PRA model. In January 2020 (Reference [36]), an F&O Closure Review was conducted for Ginna. The scope of the review included fire peer review findings. All of the findings from the 2012 fire PRA peer review were resolved. Currently, there are no open findings against the fire PRA model (Reference [36]).

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

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

The Ginna 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 Ginna Tier 1 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 scheduled review will include:

- A review of plant modifications since the last review that could impact the SSC categorization.

- A review of plant specific operating experience that could impact the SSC categorization.

- A review of the impact of the updated risk information on the categorization process results.

- A review of the importance measures used for screening in the categorization process.

- An update of the risk sensitivity study performed for the categorization.

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

The periodic monitoring requirements of the 10 CFR 50.69 process will ensure that these issues are captured and addressed at a frequency commensurate with the issue severity. The 10 CFR 50.69 periodic monitoring program includes immediate and periodic reviews, that include the requirements of the regulation, to ensure that all issues that could affect 10 CFR 50.69 categorization are addressed. The periodic monitoring process also monitors the performance and condition of categorized SSCs to ensure that the assumptions for reliability in the categorization process are maintained.

4 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 2, April 2015.
- Regulatory Guide 1.200, "An Approach for Determining the Technical Adequacy of Probabilistic Risk Assessment Results for Risk-Informed Activities," Revision 2, 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) PRA Model				
1	Model GN119A-ASM-002 Peer Reviewed Against RG 1.200 R1 in June 2009	7.5E-06	3.4E-07	2021 FPIE Application Specific Model (ASM)
Fire (FPRA) Model				
1	Model GI120AF0 Peer Reviewed Against RG 1.200 R2 in June 2012	3.8E-05	5.4E-07	2019 Fire PRA Model of Record (MOR)

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
SC-A2-01	SC-A2	CC-I	The definition of core damage documented in the Ginna-AS-Notebook; Rev. 1 Section 2.2 is consistent with the examples of measures for core damage suitable for Capability Category I as defined in NUREG/CR-4550. For Category II, Ginna could use the code-predicted core exit temperature >1,200°F for 30 min using PCTRAN (code with simplified core modeling (PWR). Review the definition of core damage and determine if PCTRAN could support the Category II core damage definition.	The Ginna PRA remains conservative with respects to the definition of core damage. For some sequences, a more realistic definition may afford some additional time for operator actions. However, over the typical loss of decay heat removal timing success criteria, the time between core uncovery and CET temperatures of 1200°F or 1800°F peak centerline is fairly small. HEPs are acknowledged as a source of uncertainty for this application. Some modest conservatism in HEPs would not adversely impact this application.

Attachment 4: External Hazards Screening

External Hazard	Screening Result		
	Screened? (Y/N)	**Screening Criterion (Note a)**	**Comment**
Aircraft Impact	Y	PS2 PS4	Acceptance criterion 1.A of Standard Review Plan 3.5.1.6 (Reference [41]) 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)**. Per UFSAR Section 2.2.2.4 (Reference [42]), the closest airport to the plant is the Williamson Flying Club Airport, a small, privately owned, general aviation facility located approximately 10 miles east-southeast of the plant. According to the Federal Aviation Administration's Air Traffic Activity System, the annual operations from this airport is less than 27,000, which is less than the 500 D^2 criteria **(PS2, PS4)**. Greater Rochester International Airport, about 25 miles southwest of the plant, is the nearest airport with scheduled commercial air service. According to the Federal Aviation

External Hazard	Screening Result		
	Screened? (Y/N)	Screening Criterion (Note a)	Comment
			Administration's Air Traffic Activity System, the annual operations from this airport is less than 85,000, which is less than the 1000 D^2 criteria **(PS2, PS4)**. Based on this review, the aircraft impact hazard is considered to be negligible.
Avalanche	Y	C3	The Ginna Nuclear Power Plant located on the south shore of Lake Ontario precludes the possibility of an avalanche. Based on this review, the Avalanche hazard can be considered to be negligible.
Biological Event	Y	C5	Per UFSAR Section 9.2.1.2.6 (Reference [42]), Lake Ontario has an infestation of zebra mussels, which makes Ginna Station's cooling systems potentially vulnerable to plugging. To control this problem, the Rochester Gas and Electric Corporation (RG&E) has installed sodium hypochlorite injection lines in the screen house inlet plenum and service water (SW) pump bays to prevent colonization of zebra mussels in the screen house bays. This is part of an overall Service Water System

External Hazard	Screening Result		
	Screened? (Y/N)	Screening Criterion (Note a)	Comment
			Reliability Optimization Program to define the techniques, equipment, methods, and responsibilities that are used to ensure the service water (SW) system performs the following functions: transfer the necessary heat from safety related equipment to the ultimate heat sink under both normal and accident conditions, provide a source of water to the preferred auxiliary feedwater system and the standby auxiliary feedwater system for decay heat removal, and support reliable and economic operation of Ginna Station. Based on this review, the Biological Event hazard can be considered to be negligible.
Coastal Erosion	Y	C1	Per UFSAR Section 2.4.4 (Reference [42]), the NRC required the placement of additional shoreline erosion protection. This protection was added to ensure minimum wave overtopping of the concrete wall fronting the plant and lower water levels in the vicinity of the screen house. The NRC performed an analysis using procedures from the Shore Protection Manual, U.S. Army Coastal Engineering Research Center of the

External Hazard	Screening Result		
	Screened? (Y/N)	**Screening Criterion (Note a)**	**Comment**
			stability and condition of the revetment fronting the plant site (Reference [43]) and concluded that if the revetment fronting the plant exists as designed, it would be capable of resisting surge flooding from Lake Ontario, and therefore, it would meet current regulatory criteria. Subsequent inspections of the revetment in November and December 1981 showed that the revetment appears to be structurally sound and stable with no evidence of major structure stability problems. Further, the inspections verified the revetment had not degraded from the original design. These revetments are monitored via the Structures Monitoring Program and Periodic Surveillance and Preventive Maintenance Programs. Therefore, it was concluded that adequate protection from surge flooding exists at Ginna Station. Based on this review, the Coastal Erosion hazard can be considered to be negligible.
Drought	Y	C5	Drought is a slowly developing hazard allowing time for orderly plant reductions, including shutdowns.

External Hazard	Screening Result		
	Screened? (Y/N)	Screening Criterion (Note a)	Comment
			Based on this review, the Drought hazard can be considered to be negligible.
External Flooding	Y	C1 PS4	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 the NRC for review on March 11, 2015 (Reference [44]). The results indicated that all flood-causing mechanisms, except Local Intense Precipitation (LIP) and combined effects River Flood that produces a probable maximum flood (PMF), were bounded by the current licensing basis (CLB) and did not pose a challenge to the plant. Peak LIP WSEs at the battery and diesel generator rooms are 255.8 ft with the buildings having a finished floor elevation of 253.5 ft. Both structures have watertight doors and seals that provide 4.5 ft protection against flood water intrusion. There are several permanently installed and normally closed doors in the Auxiliary Building and entrances to the Battery and EDG rooms. These doors are

External Hazard	Screening Result		
	Screened? (Y/N)	Screening Criterion (Note a)	Comment
			relied upon for screening the external flood hazard and therefore will be considered HSS in accordance with NEI 00-04 Figure 5-6 (**C1**). To better characterize the frequency of exceedance for the combined effects river flood risk-significant flood events, a flood-frequency study was completed on August 5, 2020 (Reference [45]). The report analyzed flooding events up to an exceedance frequency of 1E-6/yr and provided inundation mapping to show the impact to the site from a flood with an exceedance frequency of 1E-6/yr. The results show that a combined effects river flood with this exceedance frequency would not produce a water surface elevation (WSE) greater than the elevation of the stream banks on the south and east sides of the plant (**PS4**). Based on this review, the external flooding hazard can be considered to be negligible.
Extreme Wind or Tornado	Y	PS3 PS4	Based on the plant design for wind pressure and the low frequency (<1E-6/yr) of design tornadoes, a demonstrably conservative estimate of CDF associated with high wind hazard (other than tornado generated

External Hazard	Screening Result		
	Screened? (Y/N)	Screening Criterion (Note a)	Comment
			missiles) is much less than 1E-6/yr (**PS4**). Based on a plant-specific tornado missile risk analysis for Ginna (Reference [46]), more detail provided in Attachment 7, the CCDP for tornado missiles associated with design basis 150 mph (3-second gust) windspeeds is approximately 3.1E-2 and the frequency of 150 mph tornados is less than 1E-5/yr, based on the EF-scale. Therefore, tornado missiles screen (**PS3**). 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. Based on this review, the extreme wind or tornado hazard can be considered to be negligible.
Fog	Y	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 Ginna.

External Hazard	Screening Result		
	Screened? (Y/N)	**Screening Criterion (Note a)**	**Comment**
			Based on this review, the Fog hazard can be considered to be negligible.
Forest or Range Fire	Y	C4	External fires (Forest or Range Fire) 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 Ginna. Based on this review, the Forest or Range Fire hazard can be considered to be negligible.
Frost	Y	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 Ginna. Based on this review, the Frost hazard can be considered to be negligible.
Hail	Y	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 Ginna.

External Hazard	Screening Result		
	Screened? (Y/N)	Screening Criterion (Note a)	Comment
			Based on this review, the Hail hazard can be considered to be negligible.
High Summer Temperature	Y	C1 C4	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 in order to affect an orderly shutdown should temperatures exceed prescribed limits. 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**). Based on this review, the High Summer Temperature hazard can be considered to be negligible.
High Tide, Lake Level, or River Stage	Y	C5	UFSAR Appendix 2A.3 (Reference [42]) discusses Lake Ontario water level, which is under the International St. Lawrence River Board of Control with supervision and direction from the International Joint Commission of the United States and Canada.

External Hazard	Screening Result		
	Screened? (Y/N)	**Screening Criterion (Note a)**	**Comment**
			Operation and regulation criteria have been developed by the Board and its staff. The regulation plan has two sets of basic rule curves for discharge using a basic “storage equation” and supply indicators for adjusting outflows from the lake. Seasonal adjustments to the outflow curves permit storage of water in winter, spring, and early summer and the opposite in the late summer and fall, resulting in a high operating efficiency for maximum benefits to all water users. Thus, the basic water supply to the lake changes very slowly, permitting reasonably accurate forecasts and operating actions to maintain desired levels. Because of this, only minor concern is given to "short-term" supply changes, such as ice jams on the Niagara River or local winter floods (**C5**). See also External Flooding. Based on this review, the High Tide, Lake Level, or River Stage hazard can be considered to be negligible.
Hurricane	Y	C4	UFSAR 2A.3 (Reference [42]) discusses a maximum probable hurricane whose path is assumed to be similar to those of the major hurricanes of 1903, 1923, 1928, and

External Hazard	Screening Result		
	Screened? (Y/N)	**Screening Criterion (Note a)**	**Comment**
			1933, all of which entered the east coast along the Maryland-New Jersey shoreline, curving northward and over or near Lake Ontario. Maximum wind speeds in the eastern semi-circle of the hurricane would be reduced from 120 mph at the open coast to about 105 mph at the lake. Winds in the western portion of the storm would be reduced from 90 mph to about 75 mph. An average wind speed of 70 mph was used on the lake over the fetch in computing setup at the plant site. Associated rainfall was estimated at about 2 inches over the lake at the time of peak wind setup. The hurricane hazard is therefore bounded by the Extreme Wind / Tornado and External Flooding hazards for Ginna. Based on this review, the Hurricane hazard can be considered to be negligible.
Ice Cover	Y	C1 C4	The principal effects of such events would be to cause a loss of off-site power event, which is addressed for weather-related LOOP scenarios in the FPIE PRA model for Ginna (**C4**). In addition, per UFSAR Section 2.4.5 (Reference [42]), Lake Ontario seldom

External Hazard	Screening Result		
	Screened? (Y/N)	**Screening Criterion (Note a)**	**Comment**
			freezes over, but ice does occur in winter, usually along the southern and northern shores and at the northeastern end of the lake. The possibility of ice blockage of the Deer Creek discharge is considered remote. In the event of such an occurrence combined with maximum surface runoff into Deer Creek, it can be seen from Figure 2.4-4 of Reference [42] that the site topography is such as to prevent flooding the plant (**C1**). Based on this review, the Ice Cover hazard can be considered to be negligible.
Industrial or Military Facility Accident	Y	C3	Per UFSAR 2.2.2.5 (Reference [42]), Air Force Restricted Area R-5203 is located about 8 miles north of the plant site. Whenever flight activity is conducted by the Air Force within R-5203, radar surveillance is maintained by the 174th Fighter Wing, the 108th Tactical Control Group, or possibly the Cleveland Air Route Traffic Control Center. Pilots rely upon onboard navigational equipment to maintain their presence within the specified limits of the restricted area. There is also an inactive slow-speed low altitude military training route (SR-826) that passes about 6 miles west of the plant. Route SR-826 is not

External Hazard	Screening Result		
	Screened? (Y/N)	Screening Criterion (Note a)	Comment
			currently a military-controlled airspace. Acceptance criterion 1.B of Standard Review Plan 3.5.1.6 (Reference [41]) states that for military airspace, a minimum distance of 5 miles is adequate for low-level training routes, except those associated with unusual activities such as practice bombing. Air Force Restricted Area R-5203 is about 8 miles away at its closest boundary, and no unusual activities, such as bombing practice, take place. Per UFSAR 2.2.1 there is little industrial activity in the vicinity of the R. E. Ginna Nuclear Power Plant. Wayne County, where Ginna Station is located, is primarily a rural area. Typical industries in Wayne County and Monroe County are listed in Tables 2.2-1 and 2.2-2 of Reference [42]. Industrial activity is most heavily concentrated in the town of Webster, about 6 miles from the site, and consists primarily of light manufacturing. No industrial development is expected to occur in the vicinity of the Ginna site. Based on this review, the Industrial or Military Facility Accident hazard can be considered to be negligible.

External Hazard	Screening Result		
	Screened? (Y/N)	Screening Criterion (Note a)	Comment
Internal Flooding	N/A	None	The Ginna Internal Events PRA includes evaluation of risk from internal flooding events.
Internal Fire	N/A	None	The Ginna Internal Fire PRA includes evaluation of risk from internal fire events.
Landslide	Y	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	Y	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 Ginna 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 Level or River Stage	Y	C5	UFSAR Appendix 2A.3 (Reference [42]) discusses Lake Ontario water level, which is under the International St. Lawrence River Board

External Hazard	Screening Result		
	Screened? (Y/N)	Screening Criterion (Note a)	Comment
			of Control with supervision and direction from the International Joint Commission of the United States and Canada. Operation and regulation criteria have been developed by the Board and its staff. Seasonal adjustments to the outflow curves permit storage of water in winter, spring, and early summer and the opposite in the late summer and fall, resulting in a high operating efficiency for maximum benefits to all water users. Approximately 85 percent of the annual inflow to Lake Ontario comes from the upper Great Lakes with the remaining 15 percent from local drainage. Thus, the basic water supply to the lake changes very slowly (**C5**), permitting reasonably accurate forecasts and operating actions to maintain desired levels. Because of this, only minor concern is given to "short-term" supply changes, such as ice jams on the Niagara River or local winter floods. See also External Flooding. Based on this review, the Low Tide, Lake Level, or River Stage hazard can be considered to be negligible.

External Hazard	Screening Result		
	Screened? (Y/N)	**Screening Criterion (Note a)**	**Comment**
Low Winter Temperature	Y	C4 C5	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 or Satellite Impact	Y	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.
Pipeline Accident	Y	C1	Per UFSAR Section 2.2.2.2 (Reference [42]), the nearest large pipelines to the plant are a 12-in. gas line located about 6 miles southwest of the plant and a 16-in. gas line located about 10 miles south of the plant. These pipelines are far enough away to ensure pipeline accidents will not affect the safety of the plant. The gas line service to the Ginna house heating boiler and the boiler controls were reviewed and compared with

External Hazard	Screening Result		
	Screened? (Y/N)	Screening Criterion (Note a)	Comment
			National Fire Protection Association (NFPA) 85 and were found acceptable Based on this review, the Pipeline Accident hazard can be considered to be negligible.
Release of Chemicals in Onsite Storage	Y	C1	UFSAR Section 2.2.2.6 (Reference [42]) discusses onsite toxic chemicals. An onsite toxic chemical evaluation was performed by RG&E in response to the requirements of NUREG 0737, Item III.D.3.4 (Control Room Habitability) (Reference [47]). In addition, per UFSAR Section 2.2.2.6.1, sources of onsite chemical hazards were evaluated and either these chemical hazards were removed, were not likely to occur, or did not pose a threat. See also Toxic Gas (Ammonia). Based on this review, the Release of Chemicals in Onsite Storage hazard can be considered to be negligible.
River Diversion	Y	C3 C4	Per UFSAR Section 2.4.1 (Reference [42]), there are no perennial streams on the site except Deer Creek, an intermittent stream with a drainage area of about 13.3 square miles (Figure 2.1-2 Reference [42]), which enters the site from the west, passes south of the

External Hazard	Screening Result		
	Screened? (Y/N)	Screening Criterion (Note a)	Comment
			plant, and empties into the lake near the northeastern corner of the site. In addition, per UFSAR 2.4.3.4 (Reference [42]), the Ginna response to the NRC NTTF request included an evaluation of the River Diversion hazard. As stated in the UFSAR, the hazards associated with dam breaches, storm surge, seiche, tsunami, ice-induced flooding, and channel migration or diversion were determined to be implausible (**C3**) or completely bounded by other mechanisms (**C4**). Based on this review, the River Diversion hazard can be considered to be negligible.
Sand or Dust Storm	Y	C1	The plant is designed for such events. More common wind-borne dirt can occur but poses no significant risk to Ginna 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.
Seiche	Y	C3 C4	Per UFSAR 2.4.3.4 (Reference [42]), the Ginna response to the NRC NTTF request included an evaluation of the Seiche hazard.

External Hazard	Screening Result		
	Screened? (Y/N)	**Screening Criterion (Note a)**	**Comment**
			As stated in the UFSAR, the hazards associated with dam breaches, storm surge, seiche, tsunami, ice-induced flooding, and channel migration or diversion were determined to be implausible (**C3**) or completely bounded by other mechanisms (**C4**). See External Flooding. Based on this review, the Seiche hazard can be considered to be negligible.
Seismic Activity	N/A	None	See Section 3.2.3 and Figure A4-1 in this Attachment.
Snow	Y	C5	This hazard is slow to develop and can be identified via monitoring and managed via normal plant processes. Potential flooding impacts are covered under external flooding. Based on this review, the Snow hazard can be considered to be negligible.
Soil Shrink-Swell Consolidation	Y	C1 C5	The potential for this hazard is low at the site, the plant design considers this hazard, and the hazard is slow to develop and can be mitigated.

External Hazard	Screening Result		
	Screened? (Y/N)	Screening Criterion (Note a)	Comment
			Based on this review, the Soil Shrink-Swell Consolidation impact hazard can be considered to be negligible.
Storm Surge	Y	C3 C4	Per UFSAR 2.4.3.4 (Reference [42]), the Ginna response to the NRC NTTF request included an evaluation of the Storm Surge hazard. As stated in the UFSAR, the hazards associated with dam breaches, storm surge, seiche, tsunami, ice-induced flooding, and channel migration or diversion were determined to be implausible (**C3**) or completely bounded by other mechanisms (**C4**). See External Flooding. Based on this review, the Storm Surge hazard can be considered to be negligible.
Toxic Gas	Y	C1	UFSAR Section 6.4.3.2 (Reference [42]) discusses toxic gas. <u>Chlorine</u> Approximately 1.1 miles east of Ginna Station is a water treatment plant that uses chlorine to treat lake water for distribution through the Ontario water system. Additionally, 4.1 miles west of Ginna Station is a water pumping station that also uses chlorine to treat

External Hazard	Screening Result		
	Screened? (Y/N)	Screening Criterion (Note a)	Comment
			lake water. Exposure to a postulated tank rupture is mitigated by two chlorine detectors located in the outside air intake duct for the normal control room HVAC system. Upon sensing chlorine in the incoming airstream, either detector will automatically isolate the control room envelope, trip the normal HVAC system, and activate the Control Room Emergency Air Treatment System (CREATS). The exposure to control room operators is less than the 30mg/m^3 limit found in Table 1 of Regulatory Guide 1.78, Rev. 1 (Reference [48]). <u>Ammonia</u> North of the turbine building is a tank of ammonium hydroxide that is used for secondary side water treatment. Exposure to a postulated rupture of this tank is mitigated by two ammonia detectors located in the outside air intake duct for the normal control room HVAC system. Upon sensing ammonia in the incoming airstream either detector will automatically isolate the control room envelope, trip the normal HVAC system, and actuate CREATS. The calculated ammonia exposure to control room operators from this source is less than the 210 mg/m^3 limit

External Hazard	Screening Result		
	Screened? (Y/N)	Screening Criterion (Note a)	Comment
			found in Table 1 of Regulatory Guide 1.78, Rev. 1. The remaining chemicals evaluated (Halon Refrigerant, Sodium Hypochlorite, and Carbon Dioxide) are not dependent on CREATS to mitigate a postulated release and do not pose a threat to control room habitability. Based on this review, the Toxic Gas hazard can be considered to be negligible.
Transportation Accident	Y	C3 PS2	The impact of transportation accidents was evaluated in the IPEEE (Reference [49]); specifically, within the NRC GSI-156, Systematic Evaluation Program (SEP Topic 11-1.c), "Potential Hazards due to Nearby Transportation, Industrial and Military Facilities." Issues related to this topic were reviewed against the criteria of Sections 2.2.1 and 2.2.2 of the 1975 SRP, and it was determined that Ginna Station met these criteria (**PS2**). In Reference [50], Ginna submitted additional supporting information regarding this hazard that did not change the prior conclusion that the SRP criteria were met. Additionally, per UFSAR Section 2.2.1 (Reference [42]), the nearest transportation routes to the plant are

External Hazard	Screening Result		
	Screened? (Y/N)	**Screening Criterion (Note a)**	**Comment**
			Lake Road and U.S. Route 104, which pass about 1700 ft and 3.5 miles, respectively, from the plant at their closest points of approach. The highway separation distances at Ginna Station exceed the minimum distance criteria given in Regulatory Guide 1.91, Revision 1 and, therefore, provide reasonable assurance that transportation accidents resulting in explosions of truck-size shipments of hazardous materials will not have an adverse effect on the safe operation of the plant. Any large quantities of hazardous material would be shipped via U.S. Route 104, which is sufficiently distant (3.5 miles from the plant site) not to be of concern (**C3**). Based on this review, the Transportation Accident hazard can be considered to be negligible.
Tsunami	Y	C3 C4	Per UFSAR 2.4.3.4 (Reference [42]), the Ginna response to the NRC NTTF request included an evaluation of the Tsunami hazard. As stated in the UFSAR, the hazards associated with dam breaches, storm surge, seiche, tsunami, ice-induced flooding, and channel migration or diversion were determined to be implausible (**C3**) or completely bounded by other mechanisms (**C4**).

External Hazard	Screening Result		
	Screened? (Y/N)	Screening Criterion (Note a)	Comment
			See External Flooding. Based on this review, the Tsunami hazard can be considered to be negligible.
Turbine-Generated Missiles	Y	PS4	As part of the Systematic Evaluation Program (SEP Topic III-4.C), a detailed review of internally generated missile effects was conducted. Per UFSAR Section 3.5.1.2 (Reference [42]), the probability of turbine high trajectory missiles striking the safety-related systems is obtained by multiplying the conservatively estimated turbine failure and missile ejection rate, 10^{-4} per yr, by the strike probability density per turbine failure, 10^{-7} per ft^2, and by the horizontal area occupied by the systems, conservatively estimated at 12,000 ft^2. The turbine failure and missile ejection rate of 10^{-4} is conservative because of the use of a historically observed turbine failure data set. Some of the reported failures involved old turbine designs and fabrication techniques that have been improved in currently produced turbines. The resulting probability of high trajectory missile strikes is found to be on the order of 10^{-7} per yr, and the total strike probability from low and high trajectory

External Hazard	Screening Result		
	Screened? (Y/N)	Screening Criterion (Note a)	Comment
			missiles is conservatively estimated to be less than 10^{-6} per yr. Based on the Figures in the SER for SEP Topic III-4.B, the NRC staff considered the overall probability of turbine missiles damaging Ginna Station and leading to consequences in excess of 10 CFR 100 exposure guidelines is acceptably low. Based on this review, the Turbine-Generated Missiles hazard can be considered to be negligible.
Volcanic Activity	Y	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	Y	C1	Per UFSAR 2A.1.2, (Reference [42]), the maximum water level to be expected in Lake Ontario at the plant site is 250.78 ft MSL. As indicated in UFSAR 2.4.7, the plant is protected from lake surges and wind-driven waves by a shoreline revetment with a top elevation of 261.0 ft MSL. Waves associated with external flooding are covered under that hazard.

External Hazard	Screening Result		
	Screened? (Y/N)	Screening Criterion (Note a)	Comment
			See External Flooding. Based on this review, the Waves hazard can be considered to be negligible.
Note a – See Attachment 5 for descriptions of the screening criteria.			

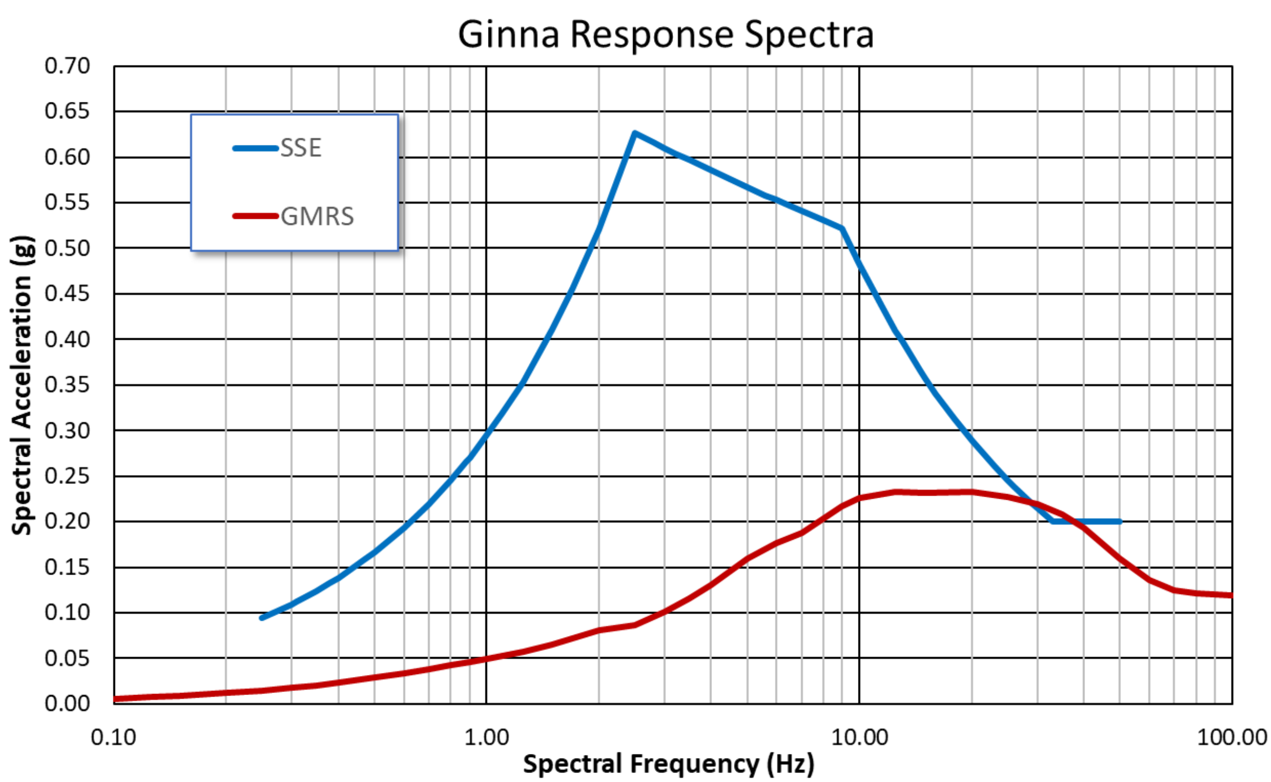

Figure A4-1: GMRS and SSE Response Spectra for Ginna

(From Reference [19], Figure 2.4-1 (GMRS) and Figure 3.1-1 (SSE)

Attachment 5: Progressive Screening Approach for Addressing External Hazards

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

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

The Ginna internal events and fire PRA models and documentation were reviewed for plant-specific modeling assumptions and related sources of uncertainty. Reference [51] and Reference [52] document sources of PRA modeling uncertainty. They identify assumptions and determine if those assumptions are related to sources of model uncertainty and characterize that uncertainty, as necessary. The identified uncertainties in Reference [51] and Reference [52] were reviewed for this application. Each PRA model includes an evaluation of the potential sources of uncertainty for the base case models using the approach that is consistent with the ASME/ANS RA-Sa-2009 [40] requirements for identification and characterization of uncertainties and assumptions. This evaluation identifies those sources of uncertainty that are important to the PRA results and may be important to PRA applications which meets the intent of steps C-1 and E-1 of NUREG-1855,Revision 1 (Reference [53]).

The results of the base PRA evaluations were reviewed to determine which potential uncertainties could impact the 10 CFR 50.69 categorization process results. This evaluation meets the intent of the screening portion of steps C-2 and E-2 of NUREG-1855, Revision 1.

Additionally, an evaluation of Level 2 Internal Events PRA model uncertainty was performed, based on the guidance in NUREG-1855 and Electric Power Research Institute (EPRI) report 1026511 (Reference [54]). The potential sources of model uncertainty in the Ginna PRA models were evaluated for the 32 Level 2 PRA topics outlined in EPRI 1026511 which is documented in Section 5 of Reference [55].

For the 10 CFR 50.69 Program, the guidance in NEI 00-04 [1] specifies sensitivity studies to be conducted for each PRA model to address key sources of uncertainty. The sensitivity studies are performed to ensure that assumptions and sources of uncertainty (e.g., human error, common cause failure, and maintenance probabilities) do not mask the SSC(s) importance. Regulatory Guide 1.174, Revision 3 (Reference [56]) cites NUREG-1855, Revision 1, as related guidance. In Section B of RG 1.174, Revision 3, the guidance acknowledges specific revisions of NUREG-1855 to include changes associated with expanding the discussion of uncertainties. The results of the evaluation of PRA model sources of uncertainty as described above are evaluated relative to the 10 CFR 50.69 application in Attachment 6 to determine if additional sensitivity evaluations are needed.

> *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 Key Assumptions/Sources of Uncertainty

The table below describes the internal events / internal flooding (IE / IF) PRA sources of model uncertainty and their impact.

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)
Loss of Offsite Power (LOOP)	The Loss of Offsite Power frequency and fail to recover offsite power probabilities are based on available industry data.	The overall approach for the LOOP frequency and failure to recover probabilities utilized is consistent with industry practice and are representative of Ginna.
Use of 24-hour mean-time-to-repair (MTTR) in support system initiating event trees.	10 CFR 50.69 analysis that involve components modeled in initiating event trees.	The use of SSIE fault trees provides an improved assessment of component importances. The of a 24 MTTR is reasonable and follows industry convention. MTTR is typically less than 24-hours.
Uncertainties associated with the assumptions and method of calculation of Human Error Probabilities (HEPs) for the Human Reliability Analysis (HRA) may introduce uncertainty.	Detailed evaluations of HEPs are performed for the risk significant human failure events (HFEs) using industry consensus methods. Mean values are used for the modeled HEPs. Uncertainty associated with the mean values can have an impact on CDF and LERF results.	Sensitivity cases performed using the base internal events PRA (HEP values of 0.0 or use of the 95th percentile value HEPs) indicate some sensitivity to human performance. Use of 95th percentile HEPs for applications is not considered realistic given the consistent use of a consensus HRA approach. The Ginna PRA model is based on industry consensus modeling approaches for its HEP calculations, so this is not considered a significant source of epistemic uncertainty. Further, as part of the 10 CFR 50.69

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)**
		evaluation upper and lower bound impacts of HEPs on classification is required.
Common cause failures	Common cause failure values are developed using available industry data.	The Ginna PRA model is based on industry consensus modeling approaches for its common cause identification and value determination. In the 10 CFR 50.69 process, a sensitivity case setting common cause factors (CCF) to 95 percentile shows some common cause sensitivity. Use of 95th percentile HEPs for applications is not considered realistic given the consistent use of a consensus CCF approach.
Core-melt arrest in-vessel	Core-melt arrest in-vessel is credited for SBO LERF sequences, using a conditional probability.	The probability of core melt arrest is not a significant contributor to risk. The Ginna LERF model is dominated by bypass events. For other accident sequences, CDF is the predominant important measure for 10 CFR 50.69.

Disposition of Key Assumptions/Sources of Uncertainty

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

Fire PRA Description	Fire PRA Sources of Uncertainty	Fire PRA Disposition
Analysis Boundary and Partitioning	This task establishes the overall spatial scope of the analysis and provides a framework for organizing the data for the analysis. The partitioning features credited are required to satisfy established industry standards.	Based on a review of the assumptions and potential sources of sources of uncertainly associated with this element it is concluded that the methodology for the Analysis Boundary and Partitioning task does not introduce any epistemic uncertainties that would affect 10 CFR 50.69.
Component Selection	This task involves the selection of components to be treated in the analysis in the context of initiating events and mitigation. The potential sources of uncertainty include those inherent in the internal events PRA model as that model provides the foundation for the FPRA.	In the context of the FPRA, one of the uncertainty issues that is unique to the analysis is related to initiating event identification. However, that impact is minimized through use of the PWROG Generic Multiple Spurious Operation (MSO) list, and the process used to identify and assess potential MSOs. As part of the Fire PRA, a small set of loads associated with uncoordinated cabling were assigned bounding routes. This was only done in the case of extremely low significance loads. A bounding sensitivity analysis was performed to measure the risk associated with this bounding routing. This concluded that there is no significant impact that would affect 10 CFR 50.69.

Fire PRA Description	Fire PRA Sources of Uncertainty	Fire PRA Disposition
Cable Selection	The selection of cables to be considered in the analysis is identified using industry guidance documents. The overall process is essentially the same as that used to perform the analyses to demonstrate compliance with 10 CFR 50.48.	Based on a review of the assumptions and potential sources of uncertainty related to this element it is concluded that the methodology for the Cable Selection task does not introduce any epistemic uncertainties that would affect 10 CFR 50.69 calculation. This concluded that there is no significant impact that would affect 10 CFR 50.69.
Qualitative Screening	Qualitative screening was performed; however, some structures (locations) were eliminated from the global analysis boundary and ignition sources deemed to have no impact on the FPRA (based on industry guidance and criteria) were excluded from the quantification based on qualitative screening criteria. The only criterion subject to uncertainty is the potential for plant trip. However, such locations would not contain any features (equipment or cables identified in the prior two tasks) and consequently are expected to have a low risk contribution.	In the event a structure (location) which could result in a plant trip was incorrectly excluded, its contribution to CDF would be small (with a CCDP commensurate with base risk). Such a location would have a negligible risk contribution to the overall FPRA. Based on a review of the assumptions and potential sources of uncertainty related to this element and the discussion above, it is concluded that the methodology for the Qualitative Screening task does not introduce any epistemic uncertainties that would affect 10 CFR 50.69.
Fire-Induced Risk Model	The internal events PRA model was updated to add fire specific initiating event structure as well as additional system logic. The methodology used is consistent with that used for	The identified source of uncertainty could result in the over-estimation of fire risk. In general, the FPRA development process would have reviewed significant fire initiating events and performed supplemental assessments to

Fire PRA Description	Fire PRA Sources of Uncertainty	Fire PRA Disposition
	the internal events PRA model development and was subjected to industry Peer Review. The developed model is applied in such a fashion that all postulated fires are assumed to generate a plant trip. This represents a source of uncertainty, as it is not necessarily clear that fires would result in a trip. In the event the fire results in damage to cables and/or equipment identified in Task 2, the PRA model includes structure to translate them into the appropriate induced initiator.	address this possible source of uncertainty. Based on a review of the assumptions and potential sources of uncertainty related to this element and the discussion above, it is concluded that the methodology for the Fire-Induced Risk Model task does not introduce any epistemic uncertainties that would affect 10 CFR 50.69.
Fire Ignition Frequencies	Fire ignition frequency is an area with inherent uncertainty. Part of this uncertainty arises due to the counting and related partitioning methodology. However, the resulting frequency is not particularly sensitive to changes in ignition source counts. The primary source of uncertainty for this task is associated with the industry generic frequency values used for the FPRA. This is because there is no specific treatment for variability among plants along with some significant conservatism in defining the	Based on a review of the assumptions and potential sources of uncertainty related to this element it is concluded that the methodology for the Fire Ignition Frequency task does not introduce any epistemic uncertainties that would affect 10 CFR 50.69. Consensus approaches are employed in the model.

Fire PRA Description	Fire PRA Sources of Uncertainty	Fire PRA Disposition
	frequencies, and their associated heat release rates. Ginna uses the ignition frequencies in NUREG-2169 (Reference [57]) along with the revised heat release rates from NUREG-2178 (Reference [58]).	
Quantitative Screening	Other than screening out potentially risk significant scenarios (ignition sources), this task is not a source of uncertainty.	The Ginna FPRA did not screen out any fire scenarios based on low CDF/LERF contribution. That is, quantified fire scenarios results are retained in the cumulative CDF/LERF. Based on the discussion above, it is concluded that the methodology for the Quantitative Screening task does not introduce any epistemic uncertainties that would affect 10 CFR 50.69.
Scoping Fire Modeling	The framework of NUREG/CR-6850 includes two tasks related to fire scenario development. These two tasks are Scoping Fire Modeling and Detailed Fire Modeling. The discussion of uncertainty for both tasks is provided in the discussion for Detailed Fire Modeling.	See the Detailed Fire Modeling discussion below.
Detailed Circuit Failure Analysis	The circuit analysis is performed using standard electrical engineering principles. However, the behavior of electrical insulation properties and the response of	Circuit analysis was performed as part of the deterministic post fire safe shutdown analysis. Refinements in the application of the circuit analysis results to the FPRA were performed on a

Fire PRA Description	Fire PRA Sources of Uncertainty	Fire PRA Disposition
	electrical circuits to fire induced failures is a potential source of uncertainty. This uncertainty is associated with the dynamics of fire and the inability to ascertain the relative timing of circuit failures. The analysis methodology assumes failures would occur in the worst possible configuration, or if multiple circuits are involved, at whatever relative timing is required to cause a bounding worst-case outcome. This results in a skewing of the risk estimates such that they are over-estimated.	case-by-case basis where the scenario risk quantification was large enough to warrant further detailed analysis. Hot short probabilities and hot short duration probabilities as defined in NUREG-7150, Volume 2, based on actual fire test data, were used in the Ginna Fire PRA. The uncertainty (conservatism) which may remain in the FPRA is associated with scenarios that do not contribute significantly to the overall fire risk. Based on a review of the assumptions and potential sources of uncertainty related to this element and the discussion above, it is concluded that the methodology for the Detailed Circuit Failure Analysis task does not introduce any epistemic uncertainties that would affect 10 CFR 50.69.
Circuit Failure Model Likelihood Analysis	One of the failure modes for a circuit (cable) given fire induced failure is a hot short. A conditional probability and a hot short duration probability are assigned using industry guidance published in NUREG-7150, Volume 2 (Reference [59]). The uncertainty values specified in NUREG-7150, Volume 2 are based on fire test data.	The use of hot short failure probability and duration probability is based on fire test data and associated consensus methodology published in NUREG/CR-7150,Volume 2. Based on a review of the assumptions and potential sources of uncertainty related to this element and the discussion above, it is concluded that the methodology for the Circuit Failure

Fire PRA Description	Fire PRA Sources of Uncertainty	Fire PRA Disposition
		Mode Likelihood Analysis task does not introduce any epistemic uncertainties that would affect 10 CFR 50.69.
Detailed Fire Modeling	The application of fire modeling technology is used in the FPRA to translate a fire initiating event into a set of consequences (fire induced failures). The performance of the analysis requires a number of key input parameters. These input parameters include the heat release rate (HRR) for the fire, the growth rate, the damage threshold for the targets, and response of plant staff (detection, fire control, fire suppression). The fire modeling methodology itself is largely empirical in some respects and consequently is another source of uncertainty. For a given set of input parameters, the fire modeling results (temperatures as a function of distance from the fire) are characterized as having some distribution (aleatory uncertainty). The epistemic uncertainty arises from the selection of the input parameters (specifically the HRR and growth rate) and how the parameters are related to the fire initiating event. While industry	Consensus modeling approach is used for the Detailed Fire Modeling. Based on a review of the assumptions and potential sources of uncertainty related to this element and the discussion above, it is concluded that the methodology for the Detailed Fire Modeling Analysis task does not introduce any epistemic uncertainties that would affect 10 CFR 50.69.

Fire PRA Description	Fire PRA Sources of Uncertainty	Fire PRA Disposition
	guidance is available, that guidance is derived from laboratory tests and may not necessarily be representative of randomly occurring events. The fire modeling results using these input parameters are used to identify a zone of influence (ZOI) for the fire and cables/equipment within that ZOI are assumed to be damaged. In general, the guidance provided for the treatment of fires is conservative and the application of that guidance retains that conservatism. The resulting risk estimates are also conservative.	
Post-Fire Human Reliability Analysis	The Human Error Probabilities (HEPs) used in the FPRA were adjusted to consider the additional challenges that may be present given a fire. The HEPs included the consideration of degradation or loss of necessary cues due to fire. Given the methodology used, the impact of any remaining uncertainties is expected to be small.	The HEPs include the consideration of degradation or loss of necessary cues due to fire. The fire risk importance measures indicate that the results are somewhat sensitive to HRA model and parameter values. The Ginna FPRA model HRA is based on industry consensus modeling approaches for its HEP calculations, so this is not considered a significant source of epistemic uncertainty. 10 CFR 50.69 applications already require assessment of the impact of operator action failure likelihood by assessing the 5% and 95%

Fire PRA Description	Fire PRA Sources of Uncertainty	Fire PRA Disposition
		percentile impact of characterization.
Seismic-Fire Interactions Assessment	Since this is a qualitative evaluation, there is no quantitative impact with respect to the uncertainty of this task.	The qualitative assessment of seismic induced fires should not be a source of model uncertainty as it is not expected to provide changes to the quantified FPRA model. Based on the discussion above, it is concluded that the methodology for the Seismic-Fire Interactions Assessment task does not introduce any epistemic uncertainties that affect 10 CFR 50.69.
Fire Risk Quantification	As the culmination of other tasks, most of the uncertainty associated with quantification has already been addressed. The other source of uncertainty is the selection of the truncation limit.	The selected truncation was confirmed to be consistent with the requirements of the PRA Standard. Based on a review of the assumptions and potential sources of uncertainty related to this element and the discussion above, it is concluded that the methodology for the Fire Risk Quantification task does not introduce any epistemic uncertainties that would affect 10 CFR 50.69.
Uncertainty and Sensitivity Analyses	This task does not introduce any new uncertainties. This task is intended to address how the fire risk assessment could be impacted by the various sources of uncertainty.	This task does not introduce any new uncertainties. This task is intended to address how the fire risk assessment could be impacted by the various sources of uncertainty. The Uncertainty and Sensitivity Analyses task does not

Fire PRA Description	Fire PRA Sources of Uncertainty	Fire PRA Disposition
		introduce any epistemic uncertainties that would affect 10 CFR 50.69.
Fire PRA Documentation	FPRA Documentation This task does not introduce any new uncertainties to the fire risk.	This task does not introduce any new uncertainties to the fire risk as it outlines documentation requirements. The methodology for the FPRA documentation task does not introduce any epistemic uncertainties that would affect 10 CFR 50.69.

Tornado Missiles - Commitment

The Ginna licensing basis for tornado missiles is described in Section 3.3.2.1.4 of the UFSAR. Further, Section 3.5.1.4 of the UFSAR documents that the facility was upgraded as part of the Structural Upgrade Program to provide adequate protection for required SSCs to perform their appropriate safety function (Reference [42]).

Subsequent to the IPEEE, Ginna performed evaluations of tornado missile protection (TMP) in order to address US NRC Regulatory Information Summary (RIS) 2015-06 (Reference [60]). Potential vulnerabilities were documented in the Ginna Tornado Missile Vulnerability Report (Reference [61]). The Ginna TMP Structural Barriers Design Analysis, DA-CE-17-001 (Reference [62]), documents the barrier upgrades and analyses to meet the design basis for TMP. Analysis DA-CE-17-001 demonstrates that the structural barriers at Ginna provide sufficiently robust missile resistance to protect safety related building and components.

An additional analysis was performed to evaluate key tornado missile barriers against the 3" pipe, weighing 78 lbs travelling at 67.6 mph (i.e., 0.4 x 169 mph, which is the windspeed associated with the 1E-6/yr tornado from NUREG/CR-4461 (Reference [63]). This analysis showed that standby auxiliary feedwater (SAFW) and B EDG structures and barriers were capable of stopping such a missile, after upgrades to several of the barriers are made (Reference [64]). This provides additional assurance that Ginna tornado missile risk is low, since key SSCs are protected against tornado missiles beyond the design basis.

The upgrades/modifications identified are (Reference [64]):

- SAFW Generator Radiator Exhaust: Replace 19W4 ¼"x2" Bar Grating with 19W4 ¼"x4" Bar Grating
- B Emergency Diesel Generator Room Air Intake: Replace 19W4 ¼"x2" Bar Grating with 19W4 ¼"x4" Bar Grating
- 'B' EDG Roof Vents: Increase anchorage capacity by expanding baseplate, increasing the size/embedment depth of anchors
- KDG08 Exhaust: Additional gussets at outside face of piping and, re-pad on outside edge of elbow
- KDG01B Exhaust: Perform field measurements to determine thickness of silencer (SDG01A) shell; upgrade as necessary