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10 CFR 50.90
10 CFR 50.69

December 26, 2019

NMP2L2716

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

Nine Mile Point Nuclear Station, Unit 2
Renewed Facility Operating License No. NPF-69
NRC Docket No. 50-410

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 Nine Mile Point Nuclear Station, Unit 2 (NMP2).

The proposed amendment would modify the NMP2 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 NMP2 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 LAR are the same as those described within the Exelon submittal of the LAR dated October 31, 2019, for Risk-Informed Technical Specifications Initiative 4b, Risk-Managed Technical Specifications (RMTS) (ADAMS Accession Number (ML 19304B653). 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 December 31, 2020, 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.

This letter contains no regulatory commitments.

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

I declare under penalty of perjury that the foregoing is true and correct. Executed on this 26th day of December 2019.

Respectfully,

David T. Andyus for

Shannon Rafferty-Czincila
Director - Licensing
Exelon Generation Company, LLC

Enclosure:

1. Evaluation of the Proposed Change

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

Enclosure
Evaluation of the Proposed Change

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LIST OF ATTACHMENTS

1 SUMMARY DESCRIPTION

The proposed amendment modifies the licensing basis to allow for the implementation of the provisions of Title 10 of the Code of Federal Regulations (10 CFR), Part 50.69, "Risk-Informed Categorization and Treatment of Structures, Systems and Components for Nuclear Power Reactors." The provisions of 10 CFR 50.69 allow adjustment of the scope of equipment subject to special treatment controls (e.g., quality assurance, testing, inspection, condition monitoring, assessment, and evaluation). For equipment determined to be of low safety significance (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 NMP2 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 [DATE], and all its subsequent associated supplements as specified in License Amendment No. [XXX] dated [DATE].
>
> Exelon will complete the items listed in Attachment 7 of Exelon letter to NRC dated [DATE] prior to implementation of 10 CFR 50.69. All issues identified in the attachment will be addressed and any associated changes will be made, focused-scope peer reviews will be performed on changes that are PRA upgrades as defined in the PRA standard(ASME/ANS RA-Sa-2009, as endorsed by RG 1.200, Revision 2), and any findings will be resolved and reflected in the PRA of record prior to implementation of the 10 CFR 50.69 categorization process.
>
> 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 LAR are the same as those described within the Exelon submittal of the LAR dated October 31, 2019 for Risk-Informed Technical Specifications Initiative 4b, Risk-Managed Technical Specifications (RMTS) (ADAMS Accession Number [ML19304B653]). 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

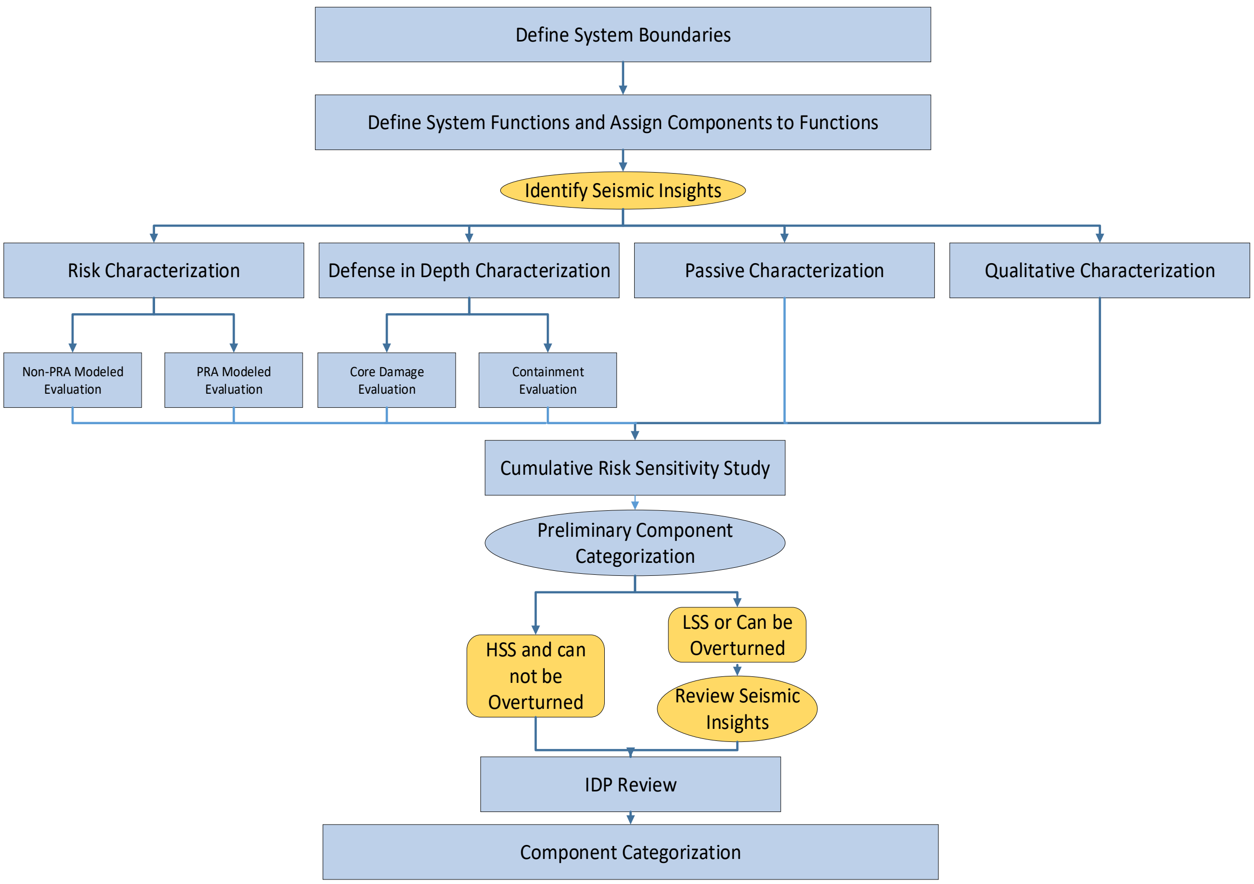
Exelon will implement the risk categorization process in accordance with NEI 00-04, Revision 0, as endorsed by Regulatory Guide (RG) 1.201, "Guidelines for Categorizing Structures, Systems, and Components in Nuclear Power Plants According to their Safety Significance" [2]. NEI 00-04 Section 1.5 states "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 3002012988 [3] approach for seismic Tier 1 sites, which includes NMP2, 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, seismic and Other External Hazards –	Component	Not Allowed	No
	Seismic –	Function/Component	Allowed [2]	No
	Shutdown – Section 5.5	Function/Component	Not Allowed	No
Defense-in-Depth	Core Damage – Section 6.1	Function/Component	Not Allowed	Yes
	Containment – Section 6.2	Component	Not Allowed	Yes
Qualitative Criteria	Considerations – Section 9.2	Function	Allowable[1]	N/A

Element	Categorization Step - NEI 00-04 Section	Evaluation Level	IDP Change HSS to LSS	Drives Associated Functions
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 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 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 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 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 NMP2 is a seismic Tier 1 (low seismic hazard) plant as defined in Reference 4, 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 [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.

- NMP2 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 3002012988 [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 October 31, 2019 ML19304B653 (Refer to Attachment 2).

- Fire Risks: Fire PRA model, as submitted to the NRC for TSTF 505 dated October 31, 2019 ML19304B653 (Refer to Attachment 2).

- Seismic Risks: EPRI Alternative Approach in EPRI 3002012988 [3] for Tier 1 plants with the additional considerations discussed in Section 3.2.3 of this LAR.

- Other External Risks (e.g., tornados, external floods): Using the IPEEE screening process as approved by NRC SE dated August 12, 1998, (TAC No. M83646). 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" [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 [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 NMP2 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 credited in this request are the same PRA models credited in the TSTF-505-A application dated October 31, 2019 ADAMS Accession Number ML19304B653 (Reference [7]).

3.2.1 Internal Events and Internal Flooding

The NMP2 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 NMP2. Attachment 2 of this enclosure identifies the applicable internal events and internal flooding PRA models.

3.2.2 Fire Hazards

The NMP2 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 NMP2. 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 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 NMP2 seismic hazard assessment, NMP2 proposes to use a risk informed graded approach that meets the requirements of 10 CFR 50.69 (b)(2) as an alternative to those listed in NEI 00-04 sections 1.5 and 5.3. This approach is specified in Reference [3] and includes additional qualitative considerations that are discussed in this section.

NMP2 meets the Tier 1 criteria for a "Low Seismic Hazard/High Seismic Margin" site. The Tier 1 criteria are as follows:

> *"Tier 1: Plants where the GMRS [Ground Motion Response Spectrum] peak acceleration is at or below approximately 0.2g or where the GMRS is below or approximately equal to the SSE [Safe Shutdown Earthquake] between 1.0 Hz and 10 Hz. Examples are shown in Figures 2-1 and 2-2. 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 3002012988 applies to the Tier 1 sites in its entirety except for the sections 2.3 (Tier 2 sites), 2.4 (Tier 3 sites), Appendix A (seismic correlation), and Appendix B (criteria for capacity-based screening).

The Tier 1 criterion (i.e. basis) in EPRI 3002012988 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 50.69 Seismic Alternative (EPRI 3002012988) will continue to compare GMRS to SSE.

The trial studies in EPRI 3002012988 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 50.69 categorization."*

The proposed categorization approach for NMP2 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 50.69 categorization process specified in NEI 00-04.

For example, the 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 integral assessment meet 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 3002012988 process for Tier 1 sites to the NMP2 10 CFR 50.69 categorization process, the IDP will be provided with the rationale for applying the EPRI 3002012988 guidance and informed of plant SSC-specific seismic insights for their consideration in the HSS/LSS deliberations.

EPRI 3002012988 recommends a risk-informed graded approach for addressing the seismic hazard in the 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 3002012998 for identifying unique seismic insights.

The seismic fragility of a 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 [8]) provide inherent seismic capacities for most SSCs that are not directly related to the site-specific seismic demand. This inherent seismic capacity is based on the non-seismic design loads (pressure, thermal, dead weight, etc.) and the required functions for the SSC. For example, a pump has a relatively high inherent seismic capacity based on its design and that same seismic capacity applies at a site with a very low demand and at a site with a very high demand. At sites with lower seismic demands such as NMP2, there is no need to perform more detailed evaluations to demonstrate the inherent seismic capacities documented in industry sources such as Reference [8]. Low seismic demand sites have lower likelihood of seismically-induced failures and lesser challenges to plant systems. This, therefore, provides the technical basis for allowing use of a graded approach for addressing seismic hazards at NMP2.

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 3002012988.

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

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

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

d. The low seismic CDF and LERF estimates lead to reasonable confidence that seismic risk contributions would allow reducing an HSS to LSS due to the 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 the other licensees and being incorporated by reference into this application is described in Case Study A (Reference [9]), Case Study C (References [10], [11]), and Case Study D (References [12], [13], [14]). 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 NMP2, 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 3002012988 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 NMP2 seismic hazard changes to medium risk (i.e., Tier 2) at some future time, NMP2 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 NMP2 meets the Tier 1 site criteria.

In response to the NRC 50.54(f) letter associated with post-Fukushima recommendations (Reference [15]), NMP2 submitted a seismic hazard screening report (Reference [16]) to the NRC. NMP2 meets the second of the Tier 1 definition criteria (GMRS < SSE in 1-10 Hz range). In addition, the maximum GMRS value for NMP2 in the 1-10 Hz range meets the first Tier 1 criterion of approximately 0.2g in Reference [3]; except for the 7-10 Hz range, where the SSE is below 0.25g. The NMP2 SSE and GMRS curves from the seismic hazard and screening response in Reference [16] are shown in Figure 1 of Attachment 4.

The NRC's staff assessment of the NMP2 seismic hazard and screening response is documented in Reference [17]. In Section 3.4 of Reference [17] the NRC concluded that the methodology used by Exelon in determining the GMRS was acceptable and that the GMRS determined by Exelon adequately characterizes the reevaluated hazard for the NMP2 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 NMP2, 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 [16], [17])
2. NTTF Recommendation 2.3 seismic walkdowns (References [18], [19])
3. NTTF Recommendation 4.2 seismic mitigation strategy assessment (S-MSA) (References [20], [21])
4. NTTF Recommendation 2.1 seismic high frequency evaluation (References [22], [23])

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 NMP2.

As an enhancement to the EPRI study results as they pertain to NMP2, the proposed NMP2 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 NMP2. 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 3002012988 study and the bases for NMP2 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 NMP2) 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 NMP2 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 NMP2 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 to assess seismic hazard risk for 50.69 with the additional reviews discussed above will ensure that reasonable confidence in the evaluations required by 10 CFR 50.69(c)(1)(iv) is achieved.

Based on the above, the Summary/Conclusion/Recommendation from Section 2.2.3 of Reference [3] applies to NMP2, i.e., NMP2 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 4, the likelihood of identifying a unique seismic insight that would cause an SSC to be designated HSS is very low. Therefore, with little to no anticipated unique seismic insights, the 50.69 categorization process using the Full Power Internal Events (FPIE) PRA and other risk evaluations along with the defense-in-depth and qualitative assessment by the IDP adequately identify the safety-significant functions and SSCs.

3.2.4 Other External Hazards

All external hazards, except for seismic, were screened for applicability to NMP2 per a plant-specific evaluation in accordance with GL 88-20 (Reference [24]) 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 NMP2 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 NMP2. 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 [25]). 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 NMP2 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 NMP2 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 NMP2 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 has 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 [26]), consistent with NRC RIS 2007-06.

The internal events PRA model was subject to a self-assessment and a full-scope peer review conducted in July 2009.

The Fire PRA model was subject to a self-assessment and a full-scope peer review conducted in June 2018.

A finding closure review was conducted on the identified PRA models in February 2019. 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 [27]) as accepted by NRC in the letter dated May 3, 2017 (ML17079A427) (Reference [28]). The results of this review have been documented and are available for NRC audit.

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

- Open items and disposition from the NMP2 RG 1.200 self-assessment.
- Open findings and disposition of the NMP2 peer reviews.

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

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

The NMP2 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 NMP2 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.

6 REFERENCES

[1] NEI 00-04, "10 CFR 50.69 SSC Categorization Guideline," Revision 0, Nuclear Energy Institute," July 2005.

[2] NRC Regulatory Guide 1.201, "Guidelines for Categorizing Structures, Systems, and Components in Nuclear Power Plants According to their Safety Significance," May 2006.

[3] Electric Power Research Institute (EPRI) 3002012988, Alternative Approaches for Addressing Seismic Risk in 10CFR50.69 Risk-Informed Categorization, July 2018.

[4] Vogtle Electric Generating Plant, Units 1 and 2 -Issuance of Amendments Re: Use of 10 CFR 50.69 (TAC NOS. ME9472 AND ME9473), December 17, 2014.

[5] NUMARC 91-06, "Guidelines for Industry Actions to Assess Shutdown Management," December 1991.

[6] ANO SER Arkansas Nuclear One, Unit 2 - Approval of Request for Alternative AN02-R&R-004, Revision 1, "Request to Use Risk-Informed Safety Classification and Treatment for Repair/Replacement Activities in Class 2 and 3 Moderate and High Energy Systems," (TAC NO. MD5250) (ML090930246), April 22, 2009.

[7] NMP2 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," October 31, 2019 (ML19304B653).

[8] EPRI NP-6041-SL, "A Methodology for Assessment of Nuclear Plant Seismic Margin, Revision 1," Electric Power Research Institute, August 1991.

[9] Plant A Seismic Probabilistic Risk Assessment Report, "Response to NRC Request Regarding Recommendation 2.1 of the Near Term Task Force Review of Insights from the Fukushima Dai-ichi Accident," August 28, 2018 (ML18240A065).

[10] Plant C, License Amendment Request to Modify Approved 10 CFR 50.69 Categorization Process, June 22, 2017 (ML17173A875).

[11] Plant C, "Issuance of Amendments Regarding Application of Seismic Probabilistic Risk Assessment Into the Previously Approved 10 CFR 50.69 Categorization Process (EPID L-2017-LLA-0248)," August 10, 2018 (ML18180A062).

[12] Seismic Probabilistic Risk Assessment for Plant D Nuclear Plant, Units 1 and 2, "Response to NRC Request for Information Pursuant to 10 CFR 50.54(f) Regarding Recommendation 2.1 of the NTTF Review of Insights from the Fukushima Dai-ichi Accident," June 30, 2017 (ML17181A485).

[13] Plant D Nuclear Plant Seismic Probabilistic Risk Assessment Supplemental Information, April 10, 2018 (ML18100A966).

[14] Plant D Nuclear Plant, Units 1 and 2, Application to Adopt 10 CFR 50.69, "Risk-informed Categorization and Treatment of Structures, Systems, and Components for Nuclear Power Reactors," November 29, 2018 (ML18334A363).

[15] U.S. Nuclear Regulatory Commission, Request for Information Pursuant to Title 10 of the Code of Federal Regulations 50.54(F) Regarding Recommendations 2.1,2.3, And 9.3, of the Near-Term Task Force Review of Insights from the Fukushima Dai-Ichi Accident, March 12, 2012 (ML12053A340).

[16] Seismic Hazard and Screening Report (CEUS Sites), "Response to NRC Request for Information Pursuant to 10 CFR 50.54(f) Regarding Recommendation 2.1 of the NTTF Review of the Fukushima Dai-ichi Accident," Attachment 3, Nine Mile Point Nuclear Station, Units 1 and 2, March 31, 2014 (ML14099A196).

[17] Nine Mile Point Nuclear Station, Units 1 And 2 - Staff 10 CFR 50.69 Assessment of Information Provided Pursuant to Title 10 Of 10 CFR 50.69 the Code of Federal Regulations Part 50, Section 50.54(f), 10 CFR 50.69 Seismic Hazard Reevaluations, Recommendation 2.1 of the 10 CFR 50.69 Near-Term Task Force Review of Insights from the Fukushima 10 CFR 50.69 Dal-ichi Accident (TAC NOS. MF3973 AND MF3974), June 16, 2015 (ML1513A660).

[18] Letter from Mary G. Korsnick to Document Control Desk (NRC), "Response to 10 CFR 50.54(f) Request for Information, Recommendation 2.3, 'Seismic'," November 27, 2012 (ML12348A086).

[19] Nuclear Regulatory Commmission Staff Assessment of Nine Mile Point Nuclear Station, Units 1 and 2 Seismic Walkdown Report, June 2, 2014 (ML1413A133).

[20] Nine Mile Point Unit 2, Mitigation Strategies Assessment (MSA) Report for the New Seismic Hazard Information - NEI 12-06, Appendix H, Revision 2, H.4.2 Path 2: GMRS < SSE with High Frequency Exceedances, May 25, 2016 (ML16147A149).

[21] Nuclear Regulatory Commission Staff Review of Mitigation Strategies Assessment Report for Nine Mile Point Unit 2, June 20, 2016 (ML16166A312).

[22] High Frequency Supplement to Seismic Hazard Screening Report, Response to NRC Request for Information Pursuant to 10 CFR 50.54(f) Regarding Recommendation 2.1 of the Near-Term Task Force Review of Insights from the Fukushima Dai-ichi Accident, December 4, 2015 (ML15338A004).

[23] Nuclear Regulatory Commission Staff Review of High Frequency Confirmation Associated with Reevaluated Seismic Hazard In Response to March 12, 2012 50.54(F) Request for Information, February 18, 2016 (ML15364A544).

[24] Generic Letter 88-20, "Individual Plant Examination of External Events (IPEEE) for Severe Accident Vulnerabilities - 10 CFR 50.54(f), Supplement 4," USNRC, June 1991..

[25] EPRI TR-1016737, "Treatment of Parameter and Model Uncertainty for Probabilistic Risk Assessments," December 2008.

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

[27] Nuclear Energy Institute (NEI) Letter to NRC, "Final Revision of Appendix X to NEI 05-04/07-12/12-16, Close-Out of Facts and Observations (F&Os)," February 21, 2017, Accession Number ML17086A431.

[28] Nuclear Regulatory Commission (NRC) Letter to Mr. Greg Krueger (NEI), "U.S. Nuclear Regulatory Commission Acceptance on Nuclear Energy Institute Appendix X to Guidance 05-04, 7-12, and 12-13, Close Out of Facts and Observations (F&Os)," May 3, 2017, Accession Number ML17079A427.

[29] Nine Mile Point Nuclear Station Unit 2, Updated Safety Analysis Report, Revision 23, October 2018.

[30] USNRC Regulatory Guide 1.70, "Standard Format and Content of Safety Analysis Reports for Nuclear Power Plants," LWR Edition, Revision 3, November 1978.

[31] USNRC Standard Review Plan, NUREG-0800 Chapter 3.5.1.6, "Aircraft Hazards," Revision 4, March 2010.

[32] USNRC Standard Review Plan, NUREG-0800, Section 2.2.3, "Evaluation of Potential Accidents," Revision 3, March 2007.

[33] ER-AA-340, "GL 89-13 Program Implementing Procedure," Revision 9.

[34] Nine Mile Point Nuclear Station, Units 1 & 2 Flood Hazard Reevaluation Report (FHRR), March 12, 2013, Docket Nos. 50-220 and 50-410.

[35] Nine Mile Point Unit 2 Updated Safety Analysis Report (USAR) Revision 20, October 2012.

[36] NUREG/CR-4461, "Tornado Climatology of the Contiguous United States," Revision 2, February 2007.

[37] Individual Plant Examination of External Events (IPEEE) Submittal, Nine Mile Station – Unit 2, SAS-TR-95-001, June 1995.

[38] Nine Mile Point Nuclear Station, Unit 2, Technical Specifications Amendment 11.

[39] Seismic Hazard and Screening Report (CEUS Sites), "Response to NRC Request for Information Pursuant to 10 CFR 50.54(f) Regarding Recommendation 2.1 of the NTTF Review of the Fukushima Dai-ichi Accident, "Attachment 3, Nine Mile Point Nuclear Station, Units 1 and 2," March 31, 2014 (ML14099A196).

[40] N2-PRA-013, Revision 1, “Nine Mile Point Nuclear Station Summary Notebook,” March 2017.

[41] N2-PRA-021.12, Nine Mile Point Nuclear Station FPRA Uncertainty and Sensitivity Notebook, Revision 1, January 2019.

[42] ASME/ANS RA-Sa-2009, "Standard for Level 1/Large Early Release Frequency Probabilistic Risk Assessment for Nuclear Power Plant Applications," Addendum A to RAS-2008, ASME, New York, NY, American Nuclear Society, La Grange Park, Illinois, February 2009.

[43] NRC NUREG-1855, "Guidance on the Treatment of Uncertainties Associated with PRAs in Risk-Informed Decision Making," Revision 1, March 2017 (ML17062A466).

[44] Electric Power Research Institute (EPRI)Technical Report TR-1026511, "Practical Guidance on the Use of PRA in Risk-Informed Applications with a Focus on the Treatment of Uncertainty," December 2012.

[45] NRC Regulatory Guide 1.174, "An Approach for Using Probabilistic Risk Assessment in Risk-Informed Decisions on Plant-Specific Changes to the Licensing Basis," Revision 3, January 2018 (ML17317A256).

[46] "Nuclear Power Plant Fire Ignition Frequency and Non-Suppression Probability Estimation Using the Updated Fire Events Database, United States Fire Event Experience Through 2009”, "NUREG-2169/EPRI 3002002936, U.S. NRC and Electric Power Research Institute, January 2015".

[47] "Refining and Characterizing Heat Release Rates from Electrical Enclosures During Fire (RACHELLE-FIRE), Volume 1: Peak Heat Release Rates and Effect of Obstructed Plume”, NUREG-2178 Vol. 1/ EPRI 3002005578, U.S. NRC and Electric Power Research Institute, Draft Report for Comment, April 2015.

[48] "Joint Assessment of Cable Damage and Quantification of Effects from Fire (JACQUE-FIRE), Volume 2: Expert Elicitation Exercise for Nuclear Power Plant Fire-Induced Electrical Circuit Failure", Final Report, NUREG/CR-7150, Vol. 1, EPRI 3002001989, U.S. NRC and Electric Power Research Institute, May 2014.

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.

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Attachment 2: Description of PRA Models Used in Categorization

Unit	Model	Baseline CDF	Baseline LERF	Comments
Full Power Internal Events (FPIE) PRA Model				
2	NM216A Peer Reviewed Against RG 1.200 R2 in July 2009	1.8E-06	2.6E-07	2016 FPIE Model of Record (MOR)
Fire (FPRA) Model				
2	NM218A Peer Reviewed Against RG 1.200 R2 in June 2018	3.1E-05	6.2E-06	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 50.69
5-1	IE-C13	Not Met	A correction factor was used with each SSIE fault tree generated initiating event frequency. These factors are applied in the fault tree logic. The purpose of the correction factor is to allow the fault tree cutsets to propagate through the model and still produce results similar to the baseline (quantification with the point estimates for the SSIEs). The purpose of the support systems fault trees is to replace the point estimate values (Bayesian update of generic data with plant experience) with plant-specific system models to better reflect the actual plant behavior. This incorporates plant-specific characteristics into the model to both provide a better plan specific estimation of the initiating event frequencies and to gain insights into the causes of these initiating events from the importance values of the components in the models. The use of these correction factors negates the benefits of using fault trees for SSIEs.	Open. Support System Initiating event fault trees can produce non-representative results when 24-hour-based basic events are adjusted to apply a year-long mission time. NMP2 employed correction factors to assure results were representative of plant experience. This finding has a small impact on PRA model results. Improved modeling without correction factors will be incorporated into the internal events PRA model prior to implementation of 50.69. Therefore, there will be no impact on 50.69 calculations.

Finding Number	Supporting Requirement(s)	Capability Category (CC)	Description	Disposition for 50.69
			These factors defeat the purpose of incorporation of plant specific characteristics in terms of the relative importance of the SSIEs and skew the importance of contributing SSCs to overall CDF. The factors used range from ~0.67 to ~5.9E-4 and have a considerable impact on the results. No justification is provided for these correction factors or their applicability to the model and there is no discussion of the differences between the SSIE fault tree frequencies and the point estimate frequencies.	
8-1	IE-C3 HR-H1	Not Met	There is no indication that a systematic review of the cut sets for each SSIE was conducted to identify the potential for recovery actions that could have prevented an upset in one of the modeled systems from proceeding to an actual initiating event. This is expected as part of ensuring that the frequencies are calculated in as realistic a manner as possible.	Open.NMP2 has employed correction factors to assure results are representative of plant experience; implicitly crediting recovery. This finding has a small impact on PRA model results. A systematic review of the cutsets comprising the SSIEs to identify any potential recovery actions and an assessment for any actions that are feasible and that would affect the frequencies

Finding Number	Supporting Requirement(s)	Capability Category (CC)	Description	Disposition for 50.69
				of associated SSIEs will be performed prior to implementation of the 50.69 program. Therefore, there will be no impact on 50.69 calculations.
8-2	IE-C10	Not Met	Some potentially important common-cause contributors are not properly assessed for initiating event SWPX. The events do not appropriately characterize failure to run over the course of a year. The common-cause events that include failures of normally operating components should be adjusted to include the failure to run per year.	Open. Support System Initiating event fault trees can produce non-representative results when 24-hour-based basic events are adjusted to apply a year-long mission time. NMP2 used a 24-hour CCF mission time to avoid dominant contribution from CCF events which have not been prevalent in plant or industry Initiating Event experience. This finding has a small impact on PRA model results. This simplification will be removed and appropriate CCF mission times mission will be used prior to implementation of the 50.69 program. Therefore, there will be no impact on 50.69 calculations.

Attachment 4: External Hazards Screening

External Hazard	Screening Result		
	Screened? (Y/N)	**Screening Criterion (Note a)**	**Comment**
Aircraft Impact	Y	PS2 PS4	Per UFSAR Section 2.2.3.1.7 [29] the nearest air corridor is approximately 22.5 km (14 mi) east of the site (Section 3.5.1.6). There are only two airfields between the 8-km (5-mi) and 24-km (15-mi) radii of the site; the Lakeside Airpark and Oswego County Airport are about 12 km (7.5 mi) and 19 km (12 mi) south of the site, respectively. The aircraft approaches to these airports are not near the plant site. The general aviation movements at these airports total approximately 1,460/yr and 19,900/yr, respectively. The annual movements are below the critical number at which a probability analysis for aircraft accidents would be required according to RG 1.70 [30]. Therefore, the probability of aircraft crashing into the site is considered to be remote, and airplane crashes need not be considered design basis events. Similarly, for helicopter operations to and from the site, the probability of a helicopter crash resulting in radiological releases in excess of 10CFR100 guidelines has been conservatively estimated to be approximately 1 x 10-6, using the methodology of NRC SRP 3.5.1.6 [31]. In accordance with SRP 2.2.3

External Hazard	Screening Result		
	Screened? (Y/N)	Screening Criterion (Note a)	Comment
			[32], additional qualitative arguments could be made which would substantially lower this probability to less than about 10-7 per year. This satisfies the requirements of RG 1.70 such that helicopter crashes need not be considered as design basis events. Based on this review, the Aircraft Impact hazard can be considered to be negligible.
Avalanche	Y	C3	The New England location of NMP2 station along Lake Ontario precludes the possibility of an avalanche. Based on this review, the Avalanche impact hazard can be considered to be negligible.
Biological Event	Y	C3 C5	Hazard is slow to develop and can be identified via monitoring and managed via standard maintenance process. Actions committed to and completed by NMP2 in response to Generic Letter 89-13 provide on-going control of biological hazards. These controls are described in Exelon procedure ER-AA-340, "GL 89-13 Program Implementing Procedure [33].

External Hazard	Screening Result		
	Screened? (Y/N)	Screening Criterion (Note a)	Comment
			Based on this review, the Biological Event impact hazard can be considered to be negligible.
Coastal Erosion	Y	C3	Per USAR Section 2.5.1.1.5, a dike was built which prevents waves from reaching unit structures thereby eliminating the hazard of shorline erosion at the site. Based on this review, the Coastal Erosion impact hazard can be considered to be negligible.
Drought	Y	C5	Drought is a slowly developing hazard allowing time for orderly plant reductions, including shutdowns. Based on this review, the Drought impact hazard can be considered to be negligible.
External Flooding	Y	C1 PS2	The evaluation of the impact of the external flooding hazard at the site was updated as a result of the NRC's post-Fukushima 50.54(f) Request for Information. The station's flood hazard reevaluation report (FHRR) was submitted to NRC for review on March 12, 2013 [34]. The results indicate that all flood causing mechanisms, except Local Intense Precipitation (LIP), are bounded by the current licensing basis (CLB) and do not pose a challenge to the plant.

External Hazard	Screening Result		
	Screened? (Y/N)	**Screening Criterion (Note a)**	**Comment**
			LIP was reevaluated and found to produce water surface elevations (WSEs) less than that of the CLB elevation of 262.5 ft at NMP Unit 2. However, the duration for which the flood is estimated to have an impact on the site was estimated to be 52.5 hours in the FHRR and the CLB estimated only 20 hours. Therefore, the impact considering the increased duration of the flooding event was evaluated. Flooding volumes were calculated based on the revised flood inundation time, in-leakage, and building drainage features. This analysis did not assume any temporary barriers were installed prior to the event starting and all doors were assumed in their normal positions. It was concluded that a LIP event will not cause enough water to enter buildings with safety-related (SR) SSCs or accumulate to a depth that affects any of the SR SSCs. Consistent with Figure 5-6 in NEI 00-04, a further evaluation was performed for all flooding mechanisms, including LIP, to determine if there are any components that participate in screened scenarios and whose failure would result in an unscreened scenario.

External Hazard	Screening Result		
	Screened? (Y/N)	Screening Criterion (Note a)	Comment
			The result was that there are several doors whose failure to be in their normal position would result in an unscreened scenario. These doors are all personnel doors that are required to be closed unless personnel are using the door. These doors would be categorized as high safety significant HSS should their associated system doors would be categorized as high safety significant HSS should their associated systems be categorized under 10 CFR 50.69. Based on this review, the External Flooding impact hazard can be considered to be negligible.
Extreme Wind or Tornado	Y	C1 PS2 PS4	Section 3.3.1 of the USAR [35] documents that the NMP2 Category I buildings are designed to withstand a fastest mile wind velocity of 90 miles per hour. USAR Section 3.3.2 and associated Table 3.2-1 documents that key equipment and structures are designed to withstand tornados with a maximum rotational velocity of 290 mph, a maximum translational velocity of 70 mph, a maximum external pressure drop of 3 psi, and a maximum rate of pressure drop of 2 psi/sec. The maximum resultant wind velocity is 360 mph. Tornado hazard frequencies used in evaluating and dispositioning

External Hazard	Screening Result		
	Screened? (Y/N)	Screening Criterion (Note a)	Comment
			potential tornado vulnerabilities in the IPEEE bound more current tornado hazard frequencies (i.e., from NUREG/CR-4461, Rev. 2 [36]. Based on the data in NUREG/CR-4461, Revision 2, design basis tornado wind speeds have a frequency less than 1E-7/yr at the NMP2 site. Tornado missile CDF documented in the IPEEE is less than 1E-7/yr; a review of more recent TMP evaluations concludes that Tornado missile CDF is much less than 1E-6/yr. Based on this review, the Extreme Wind or Tornado impact hazard can be considered to be negligible. There are no SSCs credited to screen high winds and tornado missiles, including passive and/or active components, other than Seismic Category I structures which are considered high safety significant (HSS) for 10 CFR 50.69 categorization.
Fog	Y	C1	The principal effects of such events (such as freezing fog) would be to cause a loss of off-site power and are addressed in the weather-related Loss of Offsite Power initiating event in the internal events PRA model for NMP2.

External Hazard	Screening Result		
	Screened? (Y/N)	Screening Criterion (Note a)	Comment
			Based on this review, the Fog impact hazard can be considered to be negligible.
Forest or Range Fire	Y	C3	Forest fires were screened in the IPEEE [37]. The site landscaping and lack of forestation prevent such fires from posing a threat to NMP2 station. Per USAR 2.2.3.1.4, the site is sufficiently cleared in areas adjacent to the plant that forest or brush fires pose no safety hazards. Based on this review, the Forest or Range Fire impact hazard can be considered to be negligible.
Frost	Y	C1	The principal effects of such events would be to cause a loss of off-site power and are addressed in the weather-related Loss of Offsite Power initiating event in the internal events PRA model for NMP2. Based on this review, the Frost impact hazard can be considered to be negligible.
Hail	Y	C1	The principal effects of such events would be to cause a loss of off-site power and are addressed in the weather-related Loss of Offsite Power initiating event in the internal events PRA model for NMP2. Based on this review, the Hail impact hazard can be considered to be negligible.

External Hazard	Screening Result		
	Screened? (Y/N)	Screening Criterion (Note a)	Comment
High Summer Temperature	Y	C1 C5	The principal effects of such events would result in elevated lake temperatures which are monitored by station personel. Should the ultimate heat sink temperature exceed the NMP2 Technical Specification 3.7.1 temperature limit [38], an orderly shutdown would be initiated. Based on this review, the High Summer Temperature impact hazard can be considered to be negligible.
High Tide, Lake Level, or River Stage	Y	C3 C4 C5	Per USAR 2.4.1, dams on the St. Lawrence River, the outlet to Lake Ontario, under the authority of the International St. Lawrence River Board of Control, are used to regulate the lake level. The low limit is set for el 74.37 m (244 ft) on April 1 and is maintained at or above that elevation during the entire navigation season (April 1 to November 30). The upper limit of the lake level is el 75.59 m (248 ft). Per USAR 2.4.4, potential dam failures have been considered in plant design. Tide magnitudes amount to less than 2.5 cm (1 inch). Based on this review, the High Tide, Lake Level, or River Stage impact hazard can be considered to be negligible.

External Hazard	Screening Result		
	Screened? (Y/N)	Screening Criterion (Note a)	Comment
Hurricane	Y	C3	The location of NMP2 along Lake Ontario precludes the possibility of a hurricane. Based on this review, the Hurricane impact 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 and are addressed in the weather-related Loss of Offsite Power initiating events in the internal events PRA model for NMP2. In addition, per USAR 2.4.7, the Unit 2 design incorporates features to minimize the potential for cooling water blockage by ice. Based on this review, the Ice Cover impact hazard can be considered to be negligible.
Industrial or Military Facility Accident	Y	C1 C3 PS2	There are no chemical plants, refineries, military bases, or underground gas storage facilities within 8 km (5 mi) of the plant per USAR Section 2.2.1. There are several hazardous products or materials regularly manufactured, stored, used or transported within 5 miles of the site. There are five industrial facilities that are within 5 miles of the station per Table 2.2-3 of the USAR. Hazardous chemicals used and/or stored by manufacturers within five miles of the plant were

External Hazard	Screening Result		
	Screened? (Y/N)	Screening Criterion (Note a)	Comment
			evaluated and determined to either screen from further evaluation or were determined to meet the acceptance criteria associated with Control Room operator protection as discussed in NMP2 USAR, Section 2.2. Based on this review, the Industrial or Military Facility Accident impact hazard can be considered to be negligible.
Internal Flooding	N/A	None	The NMP2 Internal Events PRA includes evaluation of risk from internal flooding events.
Internal Fire	N/A	None	The NMP2 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 impact hazard can be considered to be negligible.
Lightning	Y	C1 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 NMP2 internal events model through the

External Hazard	Screening Result		
	Screened? (Y/N)	Screening Criterion (Note a)	Comment
			incorporation of generic and plant specific data. Based on this review, the Lightning impact hazard can be considered to be negligible.
Low Lake Level or River Stage	Y	C5	Per USAR 2.4.1, dams on the St. Lawrence River, the outlet to Lake Ontario, under the authority of the International St. Lawrence River Board of Control, are used to regulate the lake level. The low limit is set for el 74.37 m (244 ft) on April 1 and is maintained at or above that elevation during the entire navigation season (April 1 to November 30). The upper limit of the lake level is el 75.59 m (248 ft). Per USAR 2.4.4, potential dam failures have been considered in plant design. Tide magnitudes amount to less than 2.5 cm (1 inch). Based on this review, the Low Lake Level or River Stage impact hazard can be considered to be negligible.
Low Winter Temperature	Y	C1 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. At worst, the loss of off-site power events would be subsumed into the base PRA model results.

External Hazard	Screening Result		
	Screened? (Y/N)	Screening Criterion (Note a)	Comment
			Based on this review, the Low Winter Temperature impact 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 make the risk impact from such events insignificant. This hazard also was reviewed as part of the IPEEE submittal [37] and screened based on low frequency of occurrence. Based on this review, the Meteorite or Satellite Impact impact hazard can be considered to be negligible.
Pipeline Accident	Y	C1	Per USAR Section 2.2.1, there are two natural gas pipelines within 5 miles of the site. The nearest gas pipeline is over 3.2 km (2 mi) from NMP2. There is not significant hazard from explosions involving these pipelines that could interact with the plant. Based on this review, the Pipeline Accident impact hazard can be considered to be negligible.
Release of Chemicals in Onsite Storage	Y	C4 PS1	The impact of releases of hazardous materials stored on-site was evaluated in the IPEEE submittal and updated in NMP2's USAR, Section 2.2.3.1.3 Additionally, a Control Room Envelope (CRE) Habitability Program is required by plant Technical Specification 5.5.13; thus these are periodically reevaluated

External Hazard	Screening Result		
	Screened? (Y/N)	Screening Criterion (Note a)	Comment
			in accordance with the specification. Other spills were determined to have no adverse effects on operation of plant equipment. Per the CRE Habitability Program hazardous chemical evaluations have been performed for all of the chemicals stored onsite. The impact of releases of chemicals in onsite storage do not pose a risk to the site. Chemicals at the NMP2 site were analyzed in accordance with the guidance provided in RG 1.78. Most of the chemical screened due to small quantities in small containers whereas the rest of the chemicals were analyzed assuming the maximum control room intake (1300 cfm in two train pressurization mode) is unfiltered. The results showed that all the chemicals were well below the toxicity limits. Therefore, none of the chemicals post a threat to control room operators at NMP2. See also "Transportation Accidents." Based on this review, the Release of Chemicals in Onsite Storage impact hazard can be considered to be negligible.
River Diversion	Y	C3	The location of NMP2 along Lake Ontario precludes the possibility of a river diversion.

External Hazard	Screening Result		
	Screened? (Y/N)	Screening Criterion (Note a)	Comment
			Based on this review, the River Diversion impact hazard can be considered to be negligible.
Sand or Dust Storm	Y	C1 C5	The plant is designed for such events. More common wind-borne dirt can occur but poses no significant risk to NMP2 given the robust structures and protective features of the plant. Based on this review, the Sand or Dust Storm impact hazard can be considered to be negligible.
Seiche	Y	C1 C3	Per the IPEEE Table 5.4-1, the principal effects of severe weather storms (including seiche flooding) would be to cause a loss of off-site power. In addition, seiche flooding was evaluated in the USAR Section 2.4.5. Water surface setup and seiche are produced by winds and atmospheric pressure gradients. These short-term lake fluctuations are generally less than 0.6m (2 ft) in amplitude. Based on this review, the Seiche impact hazard can be considered to be negligible.
Seismic Activity	N/A	None	See Section 3.2.3 and Figure A4-1 in this Attachment.

External Hazard	Screening Result		
	Screened? (Y/N)	Screening Criterion (Note a)	Comment
Snow	Y	C1 C4 C5	This hazard is slow to develop and can be identified via monitoring and managed via normal plant processes. Potential flooding impacts covered under external flooding. Based on this review, the Snow impact 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. Based on this review, the Soil Shrink-Swell Consolidation impact hazard can be considered to be negligible.
Storm Surge	Y	C3 C4	The location of NMP2 along Lake Ontario precludes the possibility of a sea level driven storm surge. Potential flooding impacts by water levels of Lake Ontario are covered under external flooding. Based on this review, the Storm Surge impact hazard can be considered to be negligible.
Toxic Gas	Y	C4	USAR Section 2.2.3.1.3 discusses toxic gas. There is no onsite storage of chlorine; sodium hypochlorite/sodium bromide biocide system is used, thus eliminating an onsite chlorine hazard. In addition, there is no possibility of an accident that could

External Hazard	Screening Result		
	Screened? (Y/N)	Screening Criterion (Note a)	Comment
			lead to the formation of flammable clouds in the vicinity of NMP2 because (1) there is no chemical plant in the vicinity; (2) no gas pipeline passes the station that presents an explosion hazard; and (3) no liquefied gases are transported in the vicinity. Per the IPEEE, the bounding analysis showed that these accidents do not significantly contribute to the plant risk. See also Transportation Accidents. Based on this review, the Toxic Gas impact hazard can be considered to be negligible.
Transportation Accident	Y	C3 C4 PS2 PS4	The impact of transportation accidents was evaluated in the IPEEE and screened out utilizing NMP2 compliance with the SRPs. Per the UFSAR: Major transportation facilities are shown on USAR Figure 2.2-1. USAR Section 2.2.3 includes an evaluation of potential accidents, and their effects on the plant or plant operation. Types of accidents considered include explosions, flammable vapor clouds, toxic chemicals, fires, collisions with intake structures, and liquid spills. Based on a comprehensive survey of industries within a 10-km (6.2-mi) radius of Unit 2, the nearest highway on which explosive

External Hazard	Screening Result		
	Screened? (Y/N)	Screening Criterion (Note a)	Comment
			materials can be transported is Route 104, which is about 6.2 km (3.9 mi) from safety-related structures. This separation distance far exceeds the safe distance for truck traffic (approximately 548.6 m, 1,800 ft) given on Figure 1 of RG 1.91. In discussions with Conrail, it was determined that no explosive or flammable materials are transported to the Oswego terminal of the rail line between Oswego and Mexico, NY. In any event, the distance from this rail line to Unit 2 is much greater than the safe distance for rail traffic given in RG 1.91. Since the nearest commercial shipping lanes on Lake Ontario are more than 10 km (6.2 mi) from Unit 2 (according to the U.S. Coast Guard), potential explosions on a ship or barge is well beyond the radius of the peak incident pressure of 1 psi as given in RG 1.91. For the Nine Mile Point site, sources of potential toxic chemical hazards include four stationary and two transportation sources within 8 km of the site. USAR Table 2.2-7 lists the chemicals associated with each source along with their quantities and distances from the Unit 2 control room air intake. The effect of an accidental release of each of the chemicals described in

External Hazard	Screening Result		
	Screened? (Y/N)	Screening Criterion (Note a)	Comment
			the previous section on control room habitability was evaluated by calculating vapor concentrations inside the control room as a function of time following the accident. This calculation is performed using the conservative methodology outlined in NUREG-0570 and utilizing the assumptions described in RG 1.78. The results of the analysis are summarized in Table 2.2-8, which indicates that none of the toxic chemicals evaluated have the potential to incapacitate the Control Room Operators. Based on this review, the Transportation Accident impact hazard can be considered to be negligible.
Tsunami	Y	C3	The location of NMP2 along Lake Ontario precludes the possibility of a tsunami. Based on this review, the Tsunami impact hazard can be considered to be negligible.
Turbine-Generated Missiles	Y	C2 PS2	Per USAR Section 3.5.1.3, At Unit 2, the original built-up type rotor design has been replaced with a monoblock type rotor design which reduces the susceptibility to stress corrosion cracking. Due to this replacement, the probability of missile generation from the Unit 2 turbine is statistically insignificant.

External Hazard	Screening Result		
	Screened? (Y/N)	Screening Criterion (Note a)	Comment
			Based on this review, the Turbine-Generated Missiles impact hazard can be considered to be negligible.
Volcanic Activity	Y	C3	Not applicable to the site because of location (no active or dormant volcanoes located near plant site). Based on this review, the Volcanic Activity impact hazard can be considered to be negligible.
Waves	Y	C3 C4	Waves associated with external flooding are covered under that hazard. Based on this review, the Waves impact hazard can be considered to be negligible.
Note a – See Attachment 5 for descriptions of the screening criteria.			

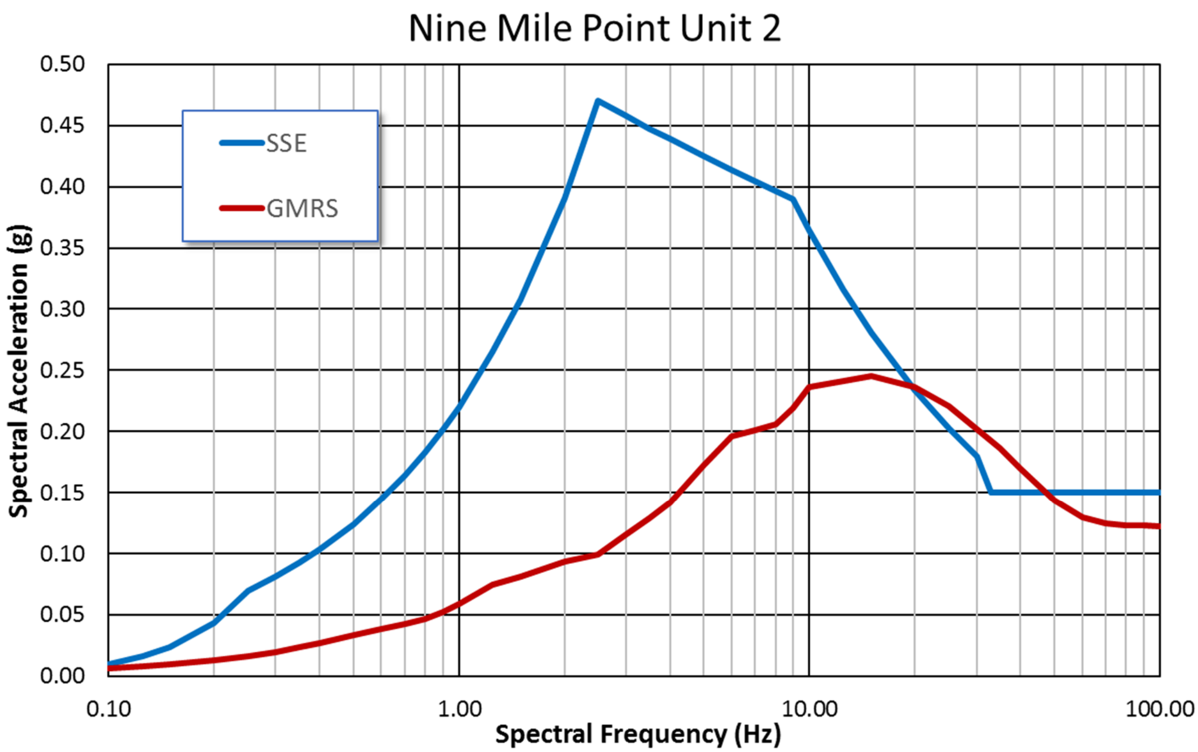

Figure A4-1: GMRS and SSE Response Spectra for NMP2
(From Reference [39], Figure 2.4-1 (GMRS) and Figure 3.1-2 (SSE)

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 NMP2 internal events and fire PRA models and documentation were reviewed for plant-specific modeling assumptions and related sources of uncertainty. Reference [40] and Reference [41] 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 [40] and Reference [41] 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 [42] 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 [43].

The results of the base PRA evaluations were reviewed to determine which potential uncertainties could impact the 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.

In order to evaluate key sources of uncertainty for the 50.69 Program application, an evaluation of Level 2 internal events PRA model uncertainty was performed, based on the guidance in NUREG-1855 (Reference [43]) and Electric Power Research Institute (EPRI) report 1026511 (Reference [44]). As described in NUREG-1855, sources of uncertainty include “parametric” uncertainties, "modeling" uncertainties, and "completeness" (or scope and level of detail) uncertainties. The potential sources of model uncertainty in the NMP2 FPRA model were evaluated for the 32 Level 2 PRA topics outlined in EPRI 1026511 (Reference [44]). It has been concluded that the Level 2 related uncertainties outlined in EPRI 1026511 do not present a significant impact on the NMP2 50.69 calculations. However, this review has highlighted the need to address time frames with the primary containment de-inerted in the calculations. This is necessary because de-inerted conditions are a configuration highlighted in importance within the Level 2 PRA results. Also note that RMAs will be developed when appropriate using insights from the Level 2 PRA model results specific to the configuration.

For the 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 [45] 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 50.69 application in Attachment 6 to determine if additional sensitivity evaluations are needed.

> *Note*: As part of the required 50.69 PRA categorization sensitivity cases directed by NEI 00-04, internal events / internal flood and fire PRA models' human error 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 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 50.69 Impact	IE / IF PRA Model Sensitivity and Disposition (50.69)
Although ECCS pumps are designed to operate under saturated conditions, uncontrolled venting as a cause of core damage is not explicitly modeled because and EOP/training has operators control containment pressure in a band and not vent in an uncontrolled way (assumed to be very unlikely that both uncontrolled vent occurs and continues up to complete loss of ECCS suction).	Scenarios for which containment heat removal systems are involved.	Uncontrolled venting has recently been added to the NMP2 PRA model (HEP ZCVUV) so that this source of uncertainty is now directly modeled in the PRA. Therefore, this will be directly included in 50.69 calculations and does not need to be separately addressed with uncertainty assessments. Because ECCS pumps are designed to operate under saturated conditions, it may now be conservative to assume failure of ECCS pumps given uncontrolled venting. However, such conditions go beyond saturated conditions in that steam will be flashing and establishing pump reliability given such a step-change in conditions would be difficult to justify. Modeling is judged to best represent the as-operated plant including explicit procedures to control containment venting pressure.
The Internal flood analysis and initiating event frequencies for spray, flood, and major	Scenarios involving internal flooding	Updated industry data is developed routinely where it is common practice to implement this new data into the model

IE / IF PRA Sources of Assumption/ Uncertainty	IE / IF PRA 50.69 Impact	IE / IF PRA Model Sensitivity and Disposition (50.69)
flood scenarios developed consistent with the EPRI methodology. One of the most important, and uncertain, inputs to an internal flooding analysis is the frequency of floods of various magnitudes (e.g., small, large, catastrophic) from various sources (e.g., clean water, untreated water, salt water, etc.). EPRI has developed some data, but the NRC has not formally endorsed its use.		during the next scheduled PRA Update. Therefore, the NMP2 PRA model will incorporate the new pipe rupture frequencies. Once implemented, this will not be a key source of uncertainty for the NMP2 50.69 Application.

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 the discussion of sources of uncertainly it is concluded that the methodology for the Analysis Boundary and Partitioning task does not introduce any epistemic uncertainties that would affect the 50.69 application.
Fire PRA 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, the uncertainty that is unique to the analysis is related to initiating event identification. However, that impact is minimized through use of the BWROG Generic Multiple Spurious Operation (MSO) list and the process used to identify and assess potential MSOs. Based on the discussion of sources of uncertainty and the discussion above, it is concluded that the methodology for the Component Selection task does not introduce any epistemic uncertainties that would affect the 50.69 application.
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 the discussion of sources of uncertainty it is concluded that the methodology for the Cable Selection task does not introduce any epistemic uncertainties that would affect the 50.69 application.
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	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.

Fire PRA Description	Fire PRA Sources of Uncertainty	Fire PRA Disposition
	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.	Based on the discussion of sources of uncertainty and the discussion above, it is concluded that the methodology for the Qualitative Screening task does not introduce any epistemic uncertainties that would affect the 50.69 application.
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 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.	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 address this possible source of uncertainty. Based on the discussion of sources of uncertainty 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 the 50.69 application.
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	Based on the discussion of sources of uncertainty, it is concluded that the methodology for the Fire Ignition Frequency task does not introduce any epistemic uncertainties that would affect the 50.69 application. Consensus approaches are employed in the model.

Fire PRA Description	Fire PRA Sources of Uncertainty	Fire PRA Disposition
	for the FPRA. This is because there is no specific treatment for variability among plants along with some significant conservatism in defining the frequencies, and their associated heat release rates. NMP2 uses the ignition frequencies in NUREG-2169 (Reference [46]) along with the revised heat release rates from NUREG - 2178 (Reference [47]).	
Quantitative Screening	Other than screening out potentially risk significant scenarios (ignition sources), this task is not a source of uncertainty.	The NMP2 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 of sources of uncertainty and the discussion above, it is concluded that the methodology for the Quantitative Screening task does not introduce any epistemic uncertainties that would affect the 50.69 application.
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 discussion for Detailed Fire Modeling.
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 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	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 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 (Reference [48]), based on actual fire test data, were used in the NMP2 Fire

Fire PRA Description	Fire PRA Sources of Uncertainty	Fire PRA Disposition
	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.	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 the discussion of sources of uncertainty 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 the 50.69 application
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 [48]). 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-7150, Volume 2. Based on the discussion of sources of uncertainty and the discussion above, it is concluded that the methodology for the Circuit Failure Mode Likelihood Analysis task does not introduce any epistemic uncertainties that would affect the 50.69 application
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	Consensus modeling approach is used for the Detailed Fire Modeling. The methodology for the Detailed Fire Modeling task does not introduce any epistemic uncertainties that would affect the 50.69 application.

Fire PRA Description	Fire PRA Sources of Uncertainty	Fire PRA Disposition
	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 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 NMP2 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. Further, as directed by NEI 00-04, the fire model human error basic events are increased to their 95th percentile and also decreased to their 5th

Fire PRA Description	Fire PRA Sources of Uncertainty	Fire PRA Disposition
		percentile values as part of the required 50.69 PRA categorization sensitivity cases. These results are capable of driving a component and respective functions HSS and therefore the uncertainty of the PRA modeled HEPs are accounted for in the 50.69 application.
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 of sources of uncertainty and the discussion above, it is concluded that the methodology for the Seismic-Fire Interactions Assessment task does not introduce any epistemic uncertainties that would affect the 50.69 application
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 the discussion of sources of uncertainty 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 the 50.69 application
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. Based on the discussion of sources of uncertainty and the discussion above, it is concluded that the methodology for the Uncertainty and Sensitivity Analyses task does not introduce any

Fire PRA Description	Fire PRA Sources of Uncertainty	Fire PRA Disposition
		epistemic uncertainties that would affect the 50.69 application
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. Based on the discussion of sources of uncertainty and the discussion above, it is concluded that the methodology for the Fire PRA Documentation task does not introduce any epistemic uncertainties that would affect the 50.69 application

Attachment 7: NMP2 50.69 PRA Implementation Items

The PRA models described within this LAR are the same as those described within the Exelon submittal of the LAR dated October 31, 2019, for Risk-Informed Technical Specifications Initiative 4b, Risk-Managed Technical Specifications (RMTS) (ADAMS Accession Number (ML19304B653).

The following list of PRA implementation items as described in Attachment 6 of the RMTS LAR will also be completed prior to implementation of the NMP2 10 CFR 50.69 risk categorization process. All issues identified will be addressed and any associated changes will be made, focused-scope peer reviews will be performed on changes that are PRA upgrades as defined in the PRA standard (ASME/ANS RA-Sa-2009, as endorsed by RG 1.200, Revision 2), and any findings will be resolved and reflected in the PRA of record prior to implementation of the 10 CFR 50.69 categorization process.

Nine Mile Point 2 PRA Implementation Items

Description	Resolution
i. Mechanical Vacuum Pump Isolation Instrumentation - One or more channels inoperable	SSCs are not modeled. The model will be updated to include these SSCs prior to implementation of the 50.69 program. The PRA Success Criteria will match the Design Success Criteria.
ii. One line with one or more suppression chamber-to-drywell vacuum breakers inoperable for opening.	The model will be updated to include this failure mode prior to implementation of the 50.69 program. The PRA Success Criteria will match the Design Success Criteria.
iii. One SW subsystem inoperable for reasons other than Conditions A and B	The success criteria are consistent with the design basis except when UHS temperature is > 82°F. The model is being updated to include this condition prior to implementation of the 50.69 program.

Nine Mile Point 2 PRA Implementation Items		
iv.	One division of intake deicer heaters inoperable.	The intake deicer heaters are not directly modeled in the PRA. The model will be updated to explicitly include these components prior to implementation of the 50.69 program.
v.	Frequency of floods of various magnitudes in the internal flooding analysis	The NMP2 PRA model will be updated to incorporate the new pipe rupture frequencies using the pipe length approach per the latest revision of EPRI TR-1013141 prior to implementation of the 50.69 program.
vi.	Nine Mile Point Unit 2 PRA Fact and Observation Independent Assessment & Focused Scope Peer Review	All open issues identified in Report 032405-RPT-01 (also shown in Attachment 3 of this application) will be addressed prior to implementation of the 50.69 program.

Attachment 8
Proposed Facility Operating License Condition Markup

Affected FOL Page

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(25) Within 14 days of the license transfers, Exelon Generation shall submit to the NRC the Nuclear Operating Services Agreement reflecting the terms set forth in the application dated August 6, 2013. Section 7.1 of the Nuclear Operating Services Agreement may not be modified in any material respect related to financial arrangements that would adversely impact the ability of the licensee to fund safety-related activities authorized by the license without the prior written consent of the Director of the Office of Nuclear Reactor Regulation.

(26) Within 10 days of the license transfers, Exelon Generation shall submit to the NRC the amended CENG Operating Agreement reflecting the terms set forth in the application dated August 6, 2013. The amended and restated Operating Agreement may not be modified in any material respect concerning decision making authority over safety, security and reliability without the prior written consent of the Director of the Office of Nuclear Reactor Regulation.

(27) At least half the members of the CENG Board of Directors must be U.S. citizens.

(28) The CENG Chief Executive Officer, Chief Nuclear Officer, and Chairman of the CENG Board of Directors must be U.S. citizens. These individuals shall have the responsibility and exclusive authority to ensure and shall ensure that the business and activities of CENG with respect to the facility's license are at all times conducted in a manner consistent with the public health and safety and common defense and security of the United States.

(29)

(30) **[Add INSERT A here]**

INSERT A

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 EPRI alternative approach described in EPRI 3002012988 for seismic risk for Tier 1 plants; as specified in License Amendment No. [XXX] dated [DATE].

Exelon will complete the implementation items listed in Attachment 7 of Exelon letter to NRC dated [DATE] prior to implementation of 10 CFR 50.69. All issues identified in the attachment will be addressed and any associated changes will be made, focused-scope peer reviews will be performed on changes that are PRA upgrades as defined in the PRA standard (ASME/ANS RA-Sa-2009, as endorsed by RG 1.200, Revision 2), and any findings will be resolved and reflected in the PRA of record prior to implementation of the 10 CFR 50.69 categorization process.

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).